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ENERGY AND WATER DEVELOPMENT APPROPRIATIONS FOR 1987

H181

ORIGINAL

HEARINGS

BEFORE A

SUBCOMMITTEE OF THE

COMMITTEE ON APPROPRIATIONS

HOUSE OF REPRESENTATIVES

NINETY-NINTH CONGRESS

SECOND SESSION

SUBCOMMITTEE ON ENERGY AND WATER DEVELOPMENT

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Staff Assistants

PART 9

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NUCLEAR SAFETY:

Department of Energy 1
Nuclear Regulatory Commission 37

Printed for the use of the Committee on Appropriations

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ENERGY AND WATER DEVELOPMENT APPROPRIATIONS FOR 1987

TUESDAY, MAY 6, 1986.

NUCLEAR REACTOR SAFETY

DEPARTMENT OF ENERGY

WITNESSES

JOSEPH F. SALGADO, UNDER SECRETARY

JAMES W. VAUGHAN, JR., ACTING ASSISTANT SECRETARY FOR NUCLEAR ENERGY

SYLVESTER R. FOLEY, JR., ASSISTANT SECRETARY FOR DEFENSE PROGRAMS

JOHN R. HUNTER, DIRECTOR, NUCLEAR ENERGY AND SURPLUS FACILITY MANAGEMENT, RICHLAND OPERATIONS OFFICE, RICHLAND, WASHINGTON

GOETZ K. OERTEL, DEPUTY ASSISTANT SECRETARY FOR SAFETY, HEALTH AND QUALITY ASSURANCE

MELVIN J. SIRES III, ASSISTANT MANAGER FOR OPERATIONS, SAVANNAH RIVER OPERATIONS OFFICE, AIKEN, SOUTH CAROLINA

J. W. THIESSEN, DEPUTY ASSOCIATE DIRECTOR FOR HEALTH AND ENVIRONMENTAL RESEARCH, OFFICE OF ENERGY RESEARCH

Mr. BEVILL. The committee will come to order.

STATEMENT OF THE CHAIRMAN

On behalf of the entire committee, I wish to thank you for attending this hearing on such short notice.

Secretary Herrington is unable to attend, but has indicated his willingness to attend another hearing or briefing next week if any members feel that is necessary.

I would like to welcome Under Secretary Joseph Salgado. This is the first time you have testified before this committee, and we are delighted to have you here.

This committee has a continuing interest in nuclear safety. This responsibility includes both the civilian nuclear industry as well as the Department of Energy's own nuclear program.

We have had several hearings on these issues already this year in connection with our review of the Fiscal Year 1987 budget. However, in light of the nuclear accident in Chernobyl, Russia, the committee is taking this opportunity to revisit the Department of Energy's and the Nuclear Regulatory Commission's nuclear safety programs.

We will hear first from the Department of Energy and then the Nuclear Regulatory Commission.

Upon completion of the testimony by the Department of Energy, the Department's witnesses are requested to remain in the room to respond to any additional questions.

At this time, Mr. Myers, I will yield to you.

Mr. **Myers**. Thank you, Mr. Chairman.

Good morning. The committee felt it necessary we give attention to our own safety procedure in light of what has happened in Russia.

This committee has always been greatly concerned about the safety of this Nation's reactors and has been proud of our record of reactor safety.

We thought we should have this hearing to see if there is any knowledge that can be gained from what happened in Russia that we might share or cause change our procedures.

We do appreciate, as the Chairman has said, your cooperation today in appearing before us.

We have not marked our bill yet. There is still time if there needs to be some change.

This is the issue that we are interested in this morning.

Thank you very much for coming.

Mr. **Bevill**. We will hear from you first, Mr. Secretary.

Opening Statement—Joseph F. Salgado

Mr. **Salgado**. Thank you, Mr. Chairman.

Mr. Chairman, thank you for the opportunity to testify before this subcommittee on the safety of the Department of Energy's research and production reactors.

I am Joseph Salgado, the Under Secretary for the Department of Energy. Accompanying me today are Admiral S. Robert Foley, the Assistant Secretary for Defense Programs and James Vaughan, the Acting Assistant Secretary for Nuclear Energy.

I would also take this opportunity to introduce Dr. Goetz Oertel, the Deputy Assistant Secretary for Safety, Health and Quality Assurance; Dr. Thiessen, Deputy Associate Director for Health and Environmental Research, Office of Energy Research; Mr. Mel Sires, Assistant Manager for Operations, Savannah River Operations Office; and Mr. John Hunter, Director of Nuclear Energy and Surplus Facility Management at the Richland Operations Office.

As you know, the Department of Energy operates a number of nuclear research and production reactors.

Although the Chernobyl disaster understandably generates concern about the safety of nuclear reactors, I want to state at the outset that we are confident in the design, safety systems and operating procedures of DOE reactors which have served in the past and continue to serve extremely well.

We are committed to continued operation of all of our reactors at safety levels in which members of this committee and the public at large can have full confidence.

Before proceeding to more detailed issues, I would like to put some perspective on DOE's safety operations.

One of Secretary Herrington's first actions over a year ago was to review the Department's health, safety and environmental programs and activities. This action was not the result of a crisis or a response to any perceived technical problems. The nuclear reactors operated by DOE are critical to our national security and to our leadership in scientific and technical research.

In ordering a review of DOE's health, safety and environmental activities, Secretary Herrington recognized that our mission to meet national security and scientific objectives carries with it a fundamental responsibility to assure worker and public health and safety.

Secretary Herrington has made it clear throughout the Department that discharging this responsibility is and will remain a top priority.

The review conducted last year noted DOE's outstanding safety record.

At the same time, the review indicated that authority for health, safety and environmental oversight was dispersed.

In response, the Secretary established the position of Assistant Secretary for Environment, Safety and Health reporting directly to his office. Several initiatives, including technical safety appraisals of all DOE's major reactor facilities were undertaken at that particular time.

In addition, the Secretary proposed a significant increase in the budget for environment, safety and health activities at DOE facilities. The budget for the Assistant Secretary, Environment, Safety and Health, grew from $46.9 million in Fiscal Year 1986 to a request of $76.1 million for Fiscal Year 1987.

TECHNICAL SAFETY APPRAISALS

Prior to the Chernobyl incident, technical safety appraisals had been scheduled at five production reactors: N-Reactor at the Hanford site in Washington, and four reactors at the Savannah River Plant in Aiken, South Carolina.

In response to the Chernobyl accident, the Secretary directed that these schedules be accelerated. As a result, a special safety appraisal of N Reactor located in Hanford, Washington, began yesterday.

Appraisals of the Savannah River reactors will be started in July rather than as previously scheduled in December.

We have elected to start with N-Reactor because it is the only one of the five that uses a graphite moderating system.

The safety appraisals review all aspects of safety, including operations, emergency planning, and, in the case of N-Reactor, graphite considerations.

Although information about the Chernobyl accident is limited, the Secretary is committed to reviewing all aspects of the accident to identify and fully consider every aspect that could be relevant to DOE reactors.

To this end, Secretary Herrington plans to invite an ad hoc group of experts to assist the Department by insuring that we examine all of the key technical issues and probe all of the potential safety issues raised by the Soviet accident.

Before closing, I would like to address a few salient points.

MULTI-TIERED SAFETY APPROACH

We have in place at all of our reactors a multi-tiered approach to safety which involves redundant systems to prevent damage to the reactor core; systems to prevent harmful radiological releases even if core damage were to occur; and effective emergency preparedness for the protection of the public and workers in the unlikely event of a large accident.

The Department has in operation eleven research and production reactors with power levels at or above 20 megawatts thermal. Six of these are research reactors and five are defense nuclear materials production reactors. The design and operations of these reactors vary; however, all use this defense in depth concept.

Other than the use of graphite, N-Reactor bears little similarity to the reactor at Chernobyl. N-Reactor fuel operates at lower temperatures than the Soviet reactor. Refueling at N-Reactor is done only with the reactor in a shut down mode; the Soviets appear to refuel while the reactor is operating.

The N-Reactor has a very sophisticated set of multiple-redundant systems to insure its safe operation.

For example, N-Reactor has: a core coolant system with redundant and independent steam supplies to the primary coolant pumps, auxiliary electric power for those pumps, diesel-driven emergency core coolant pumps, two completely independent reactor shutdown systems, a graphite cooling system with redundant pumps and coolers, an inert gas blanket over the core, and two independent sources of electric power from off site.

All nuclear reactors in the U.S.—DOE reactors as well as commercial reactors—comply with national and international standards and guidance.

The system of regulation which we have in place has its roots in the Atomic Energy Commission and has served us extremely well.

Under this system NRC regulates commercial reactors; DOE regulates the private companies and universities which operate the production and research reactors owned by the Federal Government.

The Department's nuclear safety program includes the clear assignment of responsibility and accountability for safety to DOE and contractor line managers both in Headquarters and in the field. In addition, independent safety oversight organizations at these levels assure that managers fully adhere to safety practices. Our safety systems and our regulatory framework have provided this country with a safe, effective nuclear system.

Speaking for DOE, I believe we can justifiably claim that our nuclear safety record is outstanding and excellent.

We all join President Reagan in extending our sympathies to the Russian people. The accident at Chernobyl is an extremely serious event. It should remind all of us of the grave responsibilities we bear and strengthen our resolve to continue safe, careful operation and oversight of our nuclear reactors.

The actions I have outlined here today represent a measured, rational response to the issues raised by Chernobyl. I assure you that the Department will continue to address nuclear safety as a matter of the highest priority.

Thank you for the opportunity to make my comments, sir, and I would invite any questions you and your members may have.

REACTOR SAFETY STUDIES

Mr. Bevill. How long will it take, Mr. Secretary, to complete these studies?

Mr. Salgado. The studies are broken into three separate groups, sir.

The first technical safety appraisal is targeted specifically to the graphite issue which is, of course, of great concern because of the Chernobyl incident and should be completed in a week to ten days.

The overall concept design review should be completed within 30 to 45 days. The rest of the technical safety appraisal study that is ongoing is approximately two months in length, and will be completed no later than the end of June.

The external group of experts that the Secretary will bring together this week will be mandated not only by a review of our ongoing technical appraisals, but also review the information we are able to garner from the Chernobyl incident to see if any of those potentials have any relevancy at all to the operations of N-Reactor; particularly since it is graphite moderated.

CHERNOBYL ACCIDENT

Mr. Bevill. Considering the limited amount of information that is available from the Soviet Union, could you provide the committee with a general description of the accident at Chernobyl?

Mr. Salgado. Yes, sir.

If I may preface the comments, there is a limited amount of information, number one, and there are a lot of speculations and opinions as to various causes and effects.

I would offer if, in fact, the Chairman so desires, that we would prefer to provide a classified briefing to you and members of your committee at a time convenient to you, if you so desire.

At this time, I have asked Admiral Foley to briefly overview what we know about the Chernobyl incident for you.

Admiral Foley. Good morning, Mr. Chairman.

As you said, it is quite limited information that is available to us in the public domain, but in essence, it looks to us like Chernobyl, which is one of four reactors, all of a similar type, has a roof blown off it, has one wall blown out, and these are quite thick, with debris piled up along side one of the walls.

The area where the reactor core would be is still smoking. There is evidence of some heat still remaining.

It happens that the meteorological conditions today are such that wind is blowing to the south toward the city of Chernobyl which, according to all media and reports from Europe, is being evacuated at this time almost en masse and one would presume the surrounding countryside, homes and others, are going through the same precautionary evacuation.

The levels of radiation in the area are unknown to us right now. We don't have that kind of information.

There is an awful lot we need to know in order to apply it to our own cases back here.

At the moment, it looks like a severe explosion took place.

There is obvious radioactivity in the area. There is an evacuation going on and the wind today is blowing toward Chernobyl.

Mr. **Rudd**. Mr. Chairman, I think it is important to determine whether this is a chemical explosion or a nuclear explosion.

Mr. **Bevill**. Would you comment on that, Admiral?

Admiral **Foley**. To the best of our knowledge, there is no indication whatsoever that this was a nuclear explosion.

The best information that we have and the best analysis, first accounts by all of our scientists, is that it was most likely a chemical explosion.

Mr. **Myers**. Would you yield?

Mr. **Bevill**. Mr. Myers.

Mr. **Myers**. What is the condition of the reactor? Is there likelihood of fire still existing there? Can we tell by our photographs whether graphite is still burning or not?

Admiral **Foley**. I would rather withhold any extensive dialogue on it until we can give you a classified briefing, sir.

Mr. **Myers**. That will be fine, sir.

Thank you.

Mr. **Bevill**. All right.

Mr. Secretary, can you distinguish between what is known and what is assumed; in other words, fact versus fiction?

We are getting both. Can you help us distinguish fact from fiction?

Mr. **Salgado**. I think, again, a classified briefing will help establish some parameters of fact for you, sir, but there is so much uncertainty and lack of communication and knowledge coming out of the Soviet Union itself, it will be a long time before we at the Department or we as a government are going to be able to separate fact from fiction.

There are numerous experts that are offering and tendering various opinions as to the causes of this accident.

I think that is pure speculation at this particular time. It should be noted that we are gathering certain scientific data that will probably help us, assist us in basically determining what we consider to be fact and fiction.

But that has not been assimilated, collated, or properly reviewed by an independent body at this time.

But it should be noted that the President brought together an interagency task force headed by Mr. Lee Thomas at EPA, so we are coordinating the information gathering and other activities.

But our activities vis-a-vis Europe, eastern Europe and the Soviet Union——

RUSSIAN REACTOR DESIGN

Mr. **Bevill**. I understand that the design of the Russian reactor is unique. Would you compare its design to the typical United States civilian nuclear plant?

Mr. **Salgado**. Well, I think except for the one commercial reactor I am aware of in Colorado, Fort St. Vrain reactor, all commercial reactors in the United States are light water reactors.

The reactor at Chernobyl is a graphite moderated reactor with a boiling water coolant system in it.

So, in essence, there is a major distinction between the majority of our commercial reactors and the reactor at Chernobyl.

That is the primary distinction in concept and design of the reactor itself.

Mr. **Myers**. Mr. Chairman, I wonder if he might define the difference between a graphite moderator and light water moderator?

Mr. **Salgado**. There is a graphite moderated reactor within the United States. It is a production reactor that we in the Department of Energy utilize.

I would ask Mr. Vaughan to give us a basic description of it from a technical standpoint, if it pleases the Chairman.

GRAPHITE MODERATORS

Mr. **Vaughan**. As many of you know from our discussions, the moderator is the device which slows down the neutrons in any reactor.

In each case you have fuel inside cladding, which in our light water reactors in this country is in a vessel of water, and the water acts as both a coolant and the moderator to slow down the neutrons.

The Russian reactor is a completely different design where there are individual pressure tubes with fuel in them in a series of stacked graphite blocks together, so you have 1,700 little, individual reactors grouped together.

It is a uniquely Russian design built to Russian standards and operated to their procedures.

It is just completely different from our commercial reactors.

Mr. **Myers**. Thank you.

REACTOR SAFETY

Mr. **Bevill**. Has the limited information that we have from Russia—or that the Soviet Union has made public—provided us with anything that should make this country feel that its reactors are unsafe?

Mr. **Salgado**. No, sir. It has not.

Essentially, our only reaction to the Chernobyl incident at this time is to accelerate our technical safety review to insure that not only we at the Department but you as the legislature and the country itself have the continuing confidence in our reactor safety program.

I think it should be noted that our reactor, our production reactor program, sir, in Savannah River, has been in existence for over 30 years and our N-Reactor at Hanford has 23 years of sound, safe operation.

Mr. **Bevill**. As a matter of fact, you haven't had what would be classed as a serious accident at any of the five production reactors the Department of Energy is operating, have you?

PRODUCTION REACTORS

Mr. **Salgado**. No, sir, we have not, not to my knowledge.

Mr. Bevill. Would you compare your production reactors to the typical civilian nuclear power reactor?

What is the main difference between the two?

Mr. Salgado. I will defer this to Mr. Vaughan with an overall comment.

Essentially, the purpose and design of our reactors differs from that of the commercial reactor. We are not an efficient producer of energy.

We are an efficient producer of plutonium and tritium that is necessary, because that is the design concept. That is what these reactors were built for.

The commercial reactors themselves are for the production of energy, and that is the broad concept I am aware of.

I would ask Mr. Vaughan to expand on that for you, sir.

Mr. Vaughan. The four production reactors at Savannah River are low pressure, Mr. Chairman, heavy water reactors with just a few PSI overpressure compared to the high pressure light water commercial reactor. There is very little stored energy in those Savannah River devices.

The Hanford N-Reactor is a pressurized, water cooled reactor with a graphite moderator that has separate cooling systems and with a number of typical redundant safety systems.

The fuel is loaded into the N-Reactor horizontally in lieu of being loaded vertically and has, compared to commercial reactors, a relatively short residence time in the reactor because it is being used to convert the uranium to plutonium.

Likewise, in that reactor, we do not do on-line refueling as we understand the Russians do.

CONTAINMENT VERSUS CONFINEMENT

Mr. Bevill. Would you describe the features of the production reactors and how these features eliminate the requirement for containment?

We keep hearing about containment. The five production reactors that the Department of Energy operates, have no containment. The reactors under the jurisdiction of the Nuclear Regulatory Commission have containment.

Tell us why you don't need containment, if that is the case.

Mr. Salgado. First of all, to understand the overall objective of both containment and confinement, sir, it is to insure that if there is potential for a radiological release, that off-site consequences of that release are below DOE and NRC standards.

That is the overall objective that we are trying to get to. I think we have to look at the historical context of the production reactors operated by DOE.

The Savannah River reactors were completed and constructed in 1954, 1955, I believe, and the N-Reactor in Hanford was in 1963. The design concept was developed to maintain a confinement system.

Essentially, what we are talking about is a confinement system where there is continuous filtration and removal of airborne radioactive material with subsequent release of radioactive gases within all applicable Federal public health and safety standards.

While the containment vessel itself is essentially a leak-tight structure, though as I have been advised, there could be some minor leaks of various things.

I think when we talk of containment and confinement, sir, we have to look at the full integrated safety systems that have evolved through the process, not only through the construction and design of the product but the ongoing upgrade of it.

In the overall safety design, we have redundant systems throughout the entire system.

In the N-Reactor we have cooling systems not only for the graphite but also the primary water cooling system.

We have redundant features there, safety features that are all well integrated into it.

I will ask technical people to give you a better explanation.

I think one of the most important features that I have learned about recently is that there is continuing monitoring of our entire system.

STATE OF THE ART TECHNOLOGY

We have state-of-the-art technology. It is important to note that each fuel rod assembly, each process tube, each target at Savannah River, and at N-Reactor each one of those tubes is individually monitored to insure that if there is a deviation in temperature, that it will scram the entire reactor system.

I don't believe that there is a commercial reactor or any other known reactors in the world that have that type of safety feature built in.

Basically, what we have is a redundant system that is based upon a sound safety design, sound safety operation and sound maintenance and ongoing review program of overall safety.

So, essentially, what we have when we talk of confinement is that some of the structures do not meet the large internal pressures of 130 PSIs, for example.

The confinement buildings we have, though, are not small in dimension. They are three to five feet thick concrete.

I would add that in reference to the N-Reactor, vis-a-vis the confinement/containment issue, I would ask Mr. Hunter who is from Richland, Washington, to give an explanation of the overall integrated safety systems that we have and that we sincerely feel that confinement in our safety features insures we are running a very safe and efficient plant, sir.

Mr. Hunter?

CONFINEMENT SYSTEMS

Mr. **Hunter**. Your first question had to do with why we can run N-Reactor safely without a containment system.

It is important to understand that even though we do not have a containment in the sense that is used in the commercial industry, we do have what is called a confinement system.

This consists of a very massive, concrete envelope which completely surrounds the reactor systems. The concrete is three to five feet thick. On the roof of the confiner structure we have valves, very large valves, which are steam vents.

Their function, in the event of an accident, is first of all to release the initial pressure and then to reclose so that any fission products subsequently released from the fuel are contained within the building and are vented in a controlled way through absolute filters and charcoal filters, thereby limiting any airborne releases to the environment to levels that are within established standards.

These systems are automatically triggered depending on signals which are received from the reactor systems.

If signals are received that indicate something is going very wrong, these systems will be triggered and will operate automatically to contain any possible release.

Mr. **Bevill.** So, actually, there is a form of containment.

Mr. **Hunter.** Yes, sir.

Mr. **Bevill.** For safety purposes.

Mr. **Hunter.** Yes, sir.

Mr. **Bevill.** Mr. Secretary, it seems that up to this point you are saying that the Department does not know of any way that we might make these facilities, particularly the one using the graphite system, any safer.

But to be absolutely positive or sure of the situation you have started a number of additional reviews.

Is that approximately a summary of what you have said so far?

Mr. **Salgado.** It is, sir.

The other point is our ongoing safety analysis is dynamic. It is continuing on an everyday basis, and from those things that we are learning, we seek to improve and move forward.

There are certain redundant factors we want to put into place. So, we are not in a status quo, but what we have learned from Chernobyl at this particular time has only heightened our awareness, accelerated ongoing technical appraisals and brought together an external group of experts to insure we are delving into all the potential possibilities as that drama continues to unfold.

TECHNICAL SAFETY APPRAISALS

Mr. **Bevill.** In the event you do find anything Congress should act on in the 10-day technical study, I believe you called it—this committee could appropriate funds to finance any additional safety features you may require to make them even safer.

We are going to be marking up our 1987 appropriation bill here very soon.

Mr. **Salgado.** In the contingency we would find there would be something of significant consequences we would bring it immediately to the attention of you, sir, and your committee, number one.

There would then have to be an evaluation of the dynamics of that particular increased safety envelope that we may or may not want to proceed on, but we would bring it to your attention immediately.

Mr. **Bevill.** The Chair recognizes Mr. Myers.

Mr. **Myers.** Chairman Whitten, do you have any questions?

Mr. **Whitten.** Go ahead. I appreciate your courtesy.

Mr. **Myers.** Thank you.

Gentlemen, there has been a lot of concern within and outside Congress about the Russians ignoring their responsibility under the International Atomic Energy Agency requirements.

Could you provide for the Committee a description of that agency and what requirements a member nation has to notify the agency as well as the world when an accident like this happens?

Mr. **SALGADO.** I would ask Mr. Vaughan to respond.

Mr. **VAUGHAN.** Mr. Myers, the IAEA, International Atomic Energy Agency, primarily is an organization that looks at safeguards and has a charter to assure, among other things, nonproliferation.

I am not an expert on all the rules that are required, but I have been informed that, first of all, the Soviet Union has placed some but not all of their reactors voluntarily within the monitoring of that organization.

I do not believe that under those arrangements there are precise requirements for immediately reporting and providing a defined amount of information on accidents such as this.

Frankly, it is somewhat of a fuzzy area to give you a straight answer.

BRITISH NUCLEAR ACCIDENTS

Mr. **MYERS.** There has been information that the British have experienced sometime similar—that they possibly had some kind of nuclear accident. Do you know anything about that?

Mr. **VAUGHAN.** The event that some liken to this is the problem that occurred at their Windscale reactor, which has been some number of years ago in the late 1950's. The Windscale accident back in the late 1950's is fairly well known.

That was yet a very different system, also, as I have reviewed the data. It was an air-cooled, graphite moderated reactor with metallic uranium, and there was a situation due to stored energy within the reactor that created a situation of overheating and some burning.

There have been a number of lessons learned from that which I know we have incorporated into our designs.

Mr. **MYERS.** I am familiar with that accident. They had a fire, also, in their graphite.

I heard a news story that they had a new problem. Maybe not.

Mr. **VAUGHAN.** I am not aware of a more recent reactor accident in the United Kingdom.

HUMANITARIAN ASSISTANCE

Mr. **MYERS.** The United States did offer humanitarian as well as technical assistance to the Soviets on this incident.

Have they responded to that offer?

Mr. **SALGADO.** Yes, sir.

Essentially, there was an offer made immediately upon our learning of the incident.

In fact, DOE was instrumental in drafting a good portion of the proposal offering some of our capabilities that are relatively unique, both in this country and in the world.

That offer was rejected by the Soviet Union, I believe, officially, 48 to 72 hours after our offer had been communicated.

So, it has been rejected.

Mr. MYERS. I understand that we are still willing to give technical help.

Mr. SALGADO. Yes, sir. I think we would.

I think there is an obligation of each country for sharing of things that are rather generic to all countries, and we would hope eventually, as their situation solidified, they would see fit for us to participate in an ongoing dialogue for determining causes and effects that concern all of the world.

AIRBORNE RADIATION

Mr. MYERS. I only know what I hear from the news at this point. This morning, there was a story that there were two gas clouds someplace in the northwest that did contain higher than expected radiation levels. Do we know anything more about that at this point?

Mr. SALGADO. I have asked Mr. Vaughan to prepare a brief analysis of that for you, and he would like to indicate now what we do know as our current up-to-date data on that.

Mr. VAUGHAN. As you know, the EPA is carefully monitoring that situation.

The latest report as of last night had been that they had detected the leading edge of a cloud over the west coast of the United States at about 30,000 feet, with kind of diffuse patterns of some very, very low levels of radioactivity, just slightly above background, almost minuscule levels in that cloud.

The EPA has not noted anything on the ground in the United States.

It is conceivable that if rainfall occurs from that cloud we may be able with sensitive instruments to pick something up, but certainly the levels are so low as to not be of any public health and safety concern.

That situation will continue to be monitored carefully.

Mr. SALGADO. I would indicate also, sir, that under the Interagency Task Force work—HHS, Agriculture, FEMA, and all components are monitoring as well as attempting to do some monitoring and exchange of data with overseas scientific communities.

So, we have a better feel of not only what happened, but what some of the ramifications are and how it can affect this country.

RUSSIAN AGRICULTURE

Mr. MYERS. Right after the accident there was some indication that Russian agriculture could be affected for years to come. The land could be contaminated for wheat growing and livestock and the water contaminated.

Do you have an update on that?

Mr. SALGADO. I don't believe we have sufficient data to make a definitive judgment. It is reasonable to assume, given the amount of releases that occurred at Chernobyl, there are going to be some long-term effects.

What the impacts are, I think, we probably have insufficient scientific data to render an official opinion.

Mr. MYERS. I have some other questions that either the technical people can answer later or that can be placed in the record.

Thank you very much, gentlemen.

Mr. SALGADO. Thank you, sir.

Mr. BEVILL. The Chair recognizes Mr. Whitten.

RISKS OF DEVELOPMENT OF VARIOUS ENERGY SOURCES

Mr. WHITTEN. This is a little far from the direct issue, but naturally, what has happened has the public as well as members of this committee very concerned about the future.

Recently, I read an article on the history of the invention of gun powder, TNT, steam and electricity.

Please provide a report in the record on the comparative risks that we have run with the development of various sources of power including nuclear power.

Mr. SALGADO. I would like to submit that for the record, if I could sir.

Mr. WHITTEN. Please do and include any articles by others you feel qualified.

[The information follows:]

Risks Associated With Development of Various Energy Sources

The risks associated with the development of nuclear energy for peaceful uses in the United States have been very low compared to the risks associated with the development of such energy sources as gunpowder, TNT, steam engines, and coal fired steam/electric powerplants (with their supporting fuel cycle and transportation).

While cumulative data has not been compiled, the 1985 edition of the National Safety Council publication, "Accident Facts," provides an indication of relative consequences of various activities:

Transportation	46,200
Poisoning	5,200
Falls	11,600
Fire and flames	4,800
Natural and environmental factors	1,786
Radiation	2
All other	13,600
All accidental deaths	83,188

There are about 2.5 fatalities per gigawatt electric (GWe) per year for the coal fuel cycle, due primarily to coal mining and transportation accidents. Similarly, there is about 1 fatality per GWe/year for the nuclear fuel cycle, due primarily to uranium mining-related incidents.

INTERNATIONAL AGREEMENTS

Mr. WHITTEN. As in your answer to Mr. Myers' questions, you indicated there has been difficulty with Russia about reporting on this issue.

Is there any international agreement where reports on accidents of this kind, which may affect your neighbors, would be called for. Or is that strictly up to each country?

Mr. SALGADO. I am not sure of an international agreement signed by the countries of the world.

There seems to be a human decency aspect that would indicate one should do that.

Mr. **Whitten**. We are some distance from the Soviet Union, but in view of our allies in Europe, would it be well to sponsor such an arrangement?

We seem to have difficulty reaching agreements, but that doesn't excuse us from trying.

What would be your attitude?

Mr. **Vaughan**. There is a common understanding and cooperative effort through the Nuclear Energy Agency, which is an arm of the OECD, among all the free world countries that have nuclear power plants which includes ongoing common sharing of data and experience.

That forum would surely operate to assure that happened among the free world countries.

Incident reporting is a common practice within that organization.

Mr. **Whitten**. The members of this committee are proud of the record of your operation, but in view of what has happened, it falls to us to go into the details and see, as the Chairman has said, if there is anything further we should do.

Mr. **Vaughan**. As you know, Mr. Whitten, there are three representatives of the International Atomic Energy Agency in Moscow now at the invitation of the Soviets, and we will be interested to see what information is forthcoming.

Mr. **Whitten**. Thank you very much.

I have no further questions.

Mr. **Bevill**. The Chair recognizes Mrs. Smith.

CUBAN NUCLEAR REACTOR

Mrs. **Smith**. Thank you very much, Mr. Chairman, and thank you all for being here.

Over the weekend, I heard that Cuba expects to have a nuclear reactor in five years.

From what you say, it doesn't seem that we have any international control over either the safety of these reactors or over the immediate notification in case of an accident, is that correct?

Mr. **Salgado**. I think that is correct, ma'am.

Basically, they come to these treaties voluntarily and their compliance is voluntary.

Mrs. **Smith**. Am I correct? Was the information correct? I hear Cuba expects to have one in five years?

Mr. **Salgado**. I know they have designs for one. I am not sure what their time frame is.

Mr. Vaughan?

Mr. **Vaughan**. I don't have any more specific knowledge than you read in the press.

Mrs. **Smith**. Now, I understand that we have 101 nuclear reactors, but there are 245 around the world.

Do we have any knowledge of the safety regulations for these reactors for different countries?

For example, are there any nuclear reactors in Soviet Bloc countries?

Mr. **Salgado**. The answer to the last part of your question is, yes, there are nuclear reactors in Soviet Bloc countries.

Some of them have been built by American companies. I am not quite sure of the number of countries that have joined and complied with the International safeguards and safety agreements.

Mr. Vaughan, do you?

Mr. VAUGHAN. Essentially, for a simple answer to your question, Mrs. Smith, I think we have good knowledge and understanding among the free world countries that use nuclear power.

RUSSIAN REACTOR CONTAINMENT

Mrs. SMITH. I read over the weekend that in Russia they did not have a protective envelope on this reactor. The Three Mile Island accident was contained because we did. I read they thought it was so safe they didn't need anything of this kind. Could you comment first about the accuracies of this and second, provide us your thoughts?

Mr. SALGADO. I would first indicate that from the photos that have been available in the press there did appear to be some type of concrete building surrounding the particular reactor.

Whether or not that building met any confinement criteria, as we know them or whether it met any containment criteria as we have established in the United States through the NRC, I have no idea.

Jim, do you have anything to add?

Mr. VAUGHAN. I think what we have here between the TMI situation and the Russian situation is a very different event in the sense that there was some loss of cooling and chemical reaction that in the Russian situation caused an explosion that breached the building.

The degree to which that explosion caused the building to have a hole in it and allow the radioactivity to be released is more a function of what the strength of that building is as contrasted to whether it was designed as a containment or confinement system, per se.

You are correct in that in the TMI situation there was essentially a negligible amount of radioactivity that was released off-site, and it was judged to have no significant impact to public health and safety.

It showed well the reliability of U.S. systems in a very abnormal event.

Mrs. SMITH. Just one more question—I do have more, but I want to stay within my time.

The Administration requests $98.6 million for NRC's inspection and enforcement program. Do you agree with that and what comment do you have about it?

Mr. VAUGHAN. I think it would be more appropriate to let the NRC representatives comment on that.

Mrs. SMITH. Thank you, Mr. Chairman.

Mr. BEVILL. The Chair now recognizes Mrs. Lindy Boggs.

Mrs. BOGGS. Thank you, Mr. Chairman.

I want to thank all of you for being with us.

There are reports that the Soviets have shut down all their RBMK reactors.

Are these reports correct?

Mr. Salgado. I do not have confirmation that all of their RBMK reactors have been shut down.

Mr. Vaughan?

Mr. Vaughan. We know what you read in the press, and we don't have any additional information.

I have seen that statement in the press, but I can't confirm it separately.

SIMILARITIES TO N-REACTOR

Mrs. Boggs. Since the N-Reactor is similar to the RBMK design, would it be advisable for us to shut it down until more is known about the difficulties at Chernobyl?

Mr. Salgado. In my opinion and the opinion of the Department of Energy, the answer would be no, ma'am.

I think when we say "similar", it is similar in design only in that they are graphite moderated reactors.

There are numerous differences, we believe, between the Soviet reactor and our reactor.

We are unaware of what other safety features they may or may not have had in the operation of their reactor.

We are extremely confident of the safety features, the redundant safety features, we have in the operation of our N-Reactor, and no evidence that our reactor is unsafe or being operated in an unsafe manner.

We see no reason to shut our reactor down.

NEW PRODUCTION REACTOR

Mrs. Boggs. In 1982, there was a DOE production reactor design task force. It selected a design patterned after the current Savannah River production reactors.

However, the task force did recommend that the reactor be built with a containment building.

If a new production reactor is designed with containment, should we go back and look at the old reactors to see if they should have containment added?

Mr. Salgado. Once again, ma'am, the evidence that we have, the safety record we have maintained at the Department lends no credibility to the need for containment over our existing production reactors.

Our safety systems are redundant.

They insure maximum protection for not only our workers but the communities in which we operate.

The NPR, New Production Reactor, study was commenced in 1982 and probably completed at that time, but it is an ongoing study and review.

And if the country needs a new production reactor, I am sure updated safety features will probably be added to that reactor.

CONTAINMENT BUILDINGS

Mrs. Boggs. Would it be very costly to add containment to the existing reactors?

Mr. **Salgado**. I think the magnitude of the N-Reactor, to put a containment—enclose the confinement building within a containment building would probably be impractical in a cost sense.

I think their estimates—and they are merely rough estimates—are probably anywhere between $700 million to a billion per reactor if, in fact, there was a demonstrated need to put a containment system over our confinement system that are operating safety.

Mrs. **Boggs**. Would we also have to shut down the reactor during the construction for the extra containment?

Mr. **Salgado**. I imagine we probably would. It would definitely be a very delicate situation in trying to do ongoing work over a confinement system.

N-REACTOR ELECTRICITY GENERATION

Mrs. **Boggs**. What is the electric power for the N-Reactor used for?

Mr. **Salgado**. I believe it is dispersed into the BPA grid system.

Mr. **Vaughan**. That is correct.

Mrs. **Boggs**. Are there other sources of electricity?

Mr. **Vaughan**. For the operation of the N-Reactor?

Mrs. **Boggs**. Yes.

Mr. **Salgado**. Yes, there are. There are two independent supply lines that come in off-site into the N-Reactor facility as well as redundant diesel powered systems as well as the generation of its own steam for its major coolant system.

Mrs. **Boggs**. I understand that the only completed reactor for the WPPSS project was recently shut down because its electricity was not needed.

Could the power from that plant be used to replace the N-Reactor's electricity production?

Mr. **Salgado**. I think the answer to your question is, yes, but I think the important thing to remember is that we are operating the N-Reactor at Hanford not for generating of electrical power for the Bonneville power, but it is for the production of plutonium for the national security of the United States.

Electricity happens to be a byproduct. Basically, there is power that we do sell to the BPA grid system, but the primary objective and purpose of the N-Reactor is the production of either plutonium or tritium.

AIRBORNE RADIOACTIVITY

Mrs. **Boggs**. Admiral Foley, I was noticing on this morning's news that some of the Navy aircraft in this year of the 75th anniversary of the Naval Aviation have been monitoring the radiation levels with special kinds of equipment.

Apparently, there is evidence that there is movement over the Gulf of Alaska of contamination.

Do you have any evidence of that?

Admiral **Foley**. I have read the same information, Mrs. Boggs, in the paper this morning.

In a classified briefing we can give you more information along that line.

All of the assets and facilities of the U.S. Government have been activated toward monitoring both coasts over the mainland of the United States and all the peripheral areas out there.

We have a full press going on on monitoring and acquiring any information we have and those efforts are merely part of that, ma'am.

I think to expand just for a second on that, over the years what we have seen in the Soviet Union, starting back in the Kishta incident back in the 1950's and the Cosmos satellite that fell on Canada—and it could have been designed to boost up into a higher orbit and not come down on us. The venting that the Soviets do in their testing—there is no technical reason why they have to do it, but it is just cheap.

It is a careless way to do it, and now Chernobyl indicates to me across the board that they lack a safety conscience. That is the one thing within the Department of Energy and all of our facilities that overrides everything else and that is safety.

So the efforts of the air craft and the others to acquire all this monitoring information, all that will be fed in and used in the Department of Energy and the other agencies of the government to make things safer and to learn the lessons from Chernobyl as we can gather them.

SOVIET REACTOR DESIGNS

Mrs. **Boggs**. I was interested in the reports of the shut down of the various reactors. Apparently, they haven't changed the design very much.

Admiral **Foley**. No, if you have a standard design that works and you can crank them out for economies of scale, that is fine, but clearly if you have one like Chernobyl that makes all the rest of them suspect.

We, in the Department of Energy, have a 23 year history in Richland of safe operation and over 100 reactor years down at Savannah River, so we think we have a good thing going and we want to keep it that way.

Mrs. **Boggs**. Thank you.

Thank you, Mr. Chairman.

Mr. **Bevill**. The Chair recognizes Mr. Rudd.

Mr. **Rudd**. Thank you, Mr. Chairman.

Thank you, gentlemen, for being here and thank you for your direct and enlightening testimony. It is through education that we can avoid hysteria, and that is very important.

What you have said certainly nails down the fact that we do have safety in our nuclear enterprises here in this country.

You have made it clear in every way possible what we do have. I really congratulate you on what you have done.

You have indicated that you have provided redundant systems, back-up systems, and in every way possible safety for the American people.

The radioactive cloud over the west coast, is that from the Chernobyl incident?

Mr. **Salgado**. Yes, it is.

N-REACTOR CORE CONDITION

Mr. **Rudd**. Thank you.

There is indication that even though we have all of these redundant systems and back-up systems that there may be some dangerous distortion in the graphite moderated core at Hanford and that you have elected to try to repair the reactor.

Can you comment on the condition of that reactor core and why you made the decision to repair rather than replace?

Mr. **Salgado**. There have been several factors involved.

First of all, there are some dynamics occurring with graphite and are being closely monitored.

I would reiterate again at the Hanford site where there is what you referred to as distortion or there is some movement, that each one of those process tubes is individually monitored for both flow and temperature, that if, in fact, it would be moved out of scale, went beyond the parameters, the system automatically shuts down.

Essentially, we are having engineering designs to review N-Reactor to make a determination of what is the most efficient and practical means for production from that particular reactor to meet the national security needs of this country.

And I think that one of the reasons was that if the review determines that it would be safe and efficient to modify the N-Reactor—we may go forward with that particular proposal.

I would not want to leave the impression with any member here today that the N-Reactor is being operated in any manner, shape or form that is not within safety regulations, and is an efficient, safe operation.

But we are looking at it. It is something that we are looking to the mid 1990's that we will have to address, and it is an ongoing dialogue depending upon the needs of national security as well as the design for that reactor.

NATIONAL SECURITY REQUIREMENTS

Mr. **Rudd**. Thank you for that direct enlightening statement. Does your effort to provide total safety, in any way compromise our responsibilities for national security?

Mr. **Salgado**. If, in fact, there was to be a shut-down of the reactor, then I think we would have to take a hard, fast look at our national security needs and demands, given the President's Nuclear Weapons Stockpile Memorandum.

That is of concern to us as well as the operating safety of that facility.

CONTAINMENT VERSUS CONFINEMENT

Mr. **Rudd**. Thank you.

In our commercial reactors—we refer to their "containment", and in response to questions about the Department of Energy and reactors, you refer to "confinement".

The explanation given by our friend here, indicates that there is no danger from that, but you refer to this as "confinement" rather than "containment". To me it means almost the same thing.

Can you enlighten me a little more on that?

Mr. **Salgado.** I would indicate there is, I gather, a technical, scientific difference.

If, in fact, the overall objective is to insure that the radiological release is within NRC and DOE standards, then our facility meets that objective.

There are two means and fashions of obtaining that objective, one through the containment process and one through the confinement process with our redundant safety procedures, and we feel each meet that objective.

Mr. **Rudd.** They do the same thing, in other words, so there is not really a problem between commercial and Department of Energy reactors.

Mr. **Salgado.** We do not think that our facilities are operating in a less safe manner, because we do not have containment over them vis-a-vis confinement.

We believe our safety factors and the entire integrated system of safety we utilize, utilizing the confinement system at the end of that envelope, is a safe and efficient manner in which to operate our production facilities.

NEW REACTOR RESEARCH

Mr. **Rudd.** What is the Department of Energy doing in terms of research on the next generation of reactors which may be inherently safe.

Mr. **Salgado.** We would hope to continue to develop concepts and designs, and the scientific cutting edge brings us to much safer and safer means.

I would ask Mr. Vaughan to give an analysis of where we are heading in our analysis of the program.

Mr. **Vaughan.** Mr. Rudd, you may remember my testimony before this committee regarding our program earlier this year, and we described a number of initiatives our light water reactor program conducted in concert with and complimentary to industry efforts to further enhance some of the safety features to make them potentially more forgiving to operator mistakes, which sometimes seem to catch up with us.

Beyond that, there are liquid metal cooled reactor systems and other systems which do have features in either neutronics or the thermal characteristics of the reactors, particularly in the smaller modular sizes, that would help them to be what is called passively safe or have an inherent capability to shut down even if the operators didn't take any action.

I would like to emphasize that we are always constantly trying to improve upon safety and operational features as well as state-of-the-art protective systems in an evolutionary manner.

But none of these comments that I have made should in any way imply that the commercial reactors that are operating today or the DOE reactors do not have adequate and capable redundant safety systems and operation features to protect the public, because I think the safety record of the whole nuclear power industry is excellent in terms of protecting public health and safety.

Mr. **Rudd.** I thank you. Your statement is very needed and what you are saying is most enlightening.

One thing that bothers me is that there has been a tendency for a little hysteria comparing what the Soviets do to what we do in this country with regard to nuclear energy.

It is well known that nuclear energy, if properly utilized, is safe.

You have indicated that it is safe in this country, and perhaps if we understand the difference between our two systems it might be helpful. Our system is based on respect for the individual and an open and free society. Compared to a closed society where safeguards are not taken unless world opinion forces it.

These comparisons have been made and the fact is you have said in every way possible that accident could not happen here. In your opinion, it couldn't happen here, could it?

Mr. VAUGHAN. We do not have a design of a reactor in the United States that is sufficiently similar to that accident to postulate that it could occur here.

Obviously, we will continue to evaluate whatever information comes forth to look for any lessons learned.

Mr. RUDD. Thank you very much.

Thank you, Mr. Chairman.

Mr. BEVILL. The Chair recognizes Mr. Chappell.

Mr. CHAPPELL. I have no questions.

Mr. BEVILL. The Chair recognizes Mr. Fazio.

CONTAINMENT STRUCTURES

Mr. FAZIO. Thank you, Mr. Chairman.

I have been delivered a copy of some material from the American Nuclear Energy Council, which is a group that speaks for the commercial nuclear industry in this country.

They say, in a sense to reassure the public, "The United States and other western countries place nuclear reactors into thick steel pressure vessels. These in turn, are housed in air-tight containment buildings constructed of steel-lined and steel-reinforced concrete. Containment structures in the United States are designed and built to withstand all emergency conditions, including earthquakes, tornadoes, and in some cases even the crash of large aircraft."

Mr. Salgado, are you telling me that the confinement that you have described is prepared for the kind of emergencies, unforeseen occurrences, that are described here for containment vessels around our commercial plants?

You are saying there is no difference—earthquakes, air crashes, tornadoes, are all equally protected under the confinement configuration that you use.

Is that correct?

Mr. SALGADO. Sir, given the integrated safety procedures in the system such as the individual tube monitoring system, the redundant safety features, we are saying we are meeting the objectives of a safe operating facility.

We are saying that seismically, yes, I believe our facilities have been analyzed for their seismic and have been constructed——

SEISMIC TESTS

Mr. FAZIO. All of the Department's facilities have been tested for seismic conditions and all meet the tests.

Mr. Salgado. If we are talking about all of our facilities, I cannot say.

Mr. Fazio. Tell us what you can.

Mr. Salgado. Our Savannah River reactors——

Mr. Sires. They are tested for seismic considerations and they meet those.

Mr. Hunter. Our N-Reactor is constructed to design basis for earthquake and tornado.

Mr. Salgado. As far as research reactors, test reactors, and other buildings and facilities that we have such as our national laboratories, I don't have that information, but I would be happy to provide it for the record.

Mr. Fazio. Would you also define for us what the seismic test would amount to?

Somehow we need, obviously, to gauge the degree of seismic activity that was used. I think that information would be very helpful in reaffirming the answers you have given us earlier.

Mr. Salgado. I would like to provide that for the record.

[The information follows:]

Seismic Analysis

The Department's production reactors meet the seismic and geologic design criteria of the Code of Federal Regulations, Title 10, Part 100, Appendix A. In meeting this design criteria, structures, systems, and components important to safety (including essential support facilities) have been upgraded to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components reflect appropriate natural phenomena that have been historically reported for the specific site and surrounding area.

It is impossible to seismically stress test a complete facility without in essence having a seismic event. However, all structures, systems, and components important to safety are designed and analyzed to meet the seismic criteria. In addition, components important to safety are seismically qualified (tested) by the vendors in accordance with accepted industry standards such as those promulgated by ANSI and IEEE.

Mr. Fazio. Let me ask you this—I may want to also ask people from the Nuclear Regulatory Commission.

Given the justification you have, the redundancy, the very frequent and close monitoring wouldn't it be the recommendation of the Department of Energy, in order to help the nuclear industry, that we go in the direction of developing an alternative to containment?

Shouldn't we be promoting confinements? It costs less, apparently. That might give a boost to the nuclear industry in terms of their ability to finance their future.

CONTAINMENT VERSUS CONFINEMENT

Mr. Salgado. Again, I am not a technical expert, but I would draw back to the underlying principle upon which the production reactors are not high-stored energy efficient reactors.

They produce plutonium. They use low pressure, low temperatures.

The commercial reactor produces high energy to develop a maximum capacity for energy.

Essentially, we have two basic underlying design concepts that are different than each other.

So, I would not go back and say—and I am not an expert to say—that confinement is appropriate, given the concept design of commercial light water reactors.

Mr. **Vaughan**. The only thing I would add, Mr. Fazio, is that some of the advanced passively safe, or inherently safe concepts that people are advocating, would propose to rely on those features in lieu of separate containment features.

Those designs and the data to support them are a matter of continuing review by the Department of Energy with the NRC, and the degree to which it can ultimately lead to commercial criteria is a matter that will be determined as part of the ongoing design process.

ADVANCED REACTOR CONCEPTS

Mr. **Fazio**. Could you describe some of those new types of reactors that you think might be possible to develop and operate commercially with confinement as opposed to containment?

Mr. **Vaughan**. I just described for Mr. Rudd a couple of those—the smaller, modular liquid metal cooled reactors.

You may remember that we recently ran a demonstration at EBR II in Idaho in which we literally had what was a simulated loss of cooling accident, and it shut itself down with only a 200 degree temperature rise, for example.

Mr. **Fazio**. I read that in the Wall Street Journal just two or three days ago.

Mr. **Vaughan**. That was a rather bold test, but it demonstrates what smaller reactors with certain design features can do.

Mr. **Fazio**. The HTGR at Fort St. Vrain is another example of technology that wouldn't require the large containment vessel.

Mr. **Vaughan**. There is a difference between the Fort St. Vrain reactor and the smaller modular high temperature gas cooled reactors that are being advocated by that design community today.

There are a number of heat holding capabilities of the graphite, and the fact that they are cooled with an inert gas is quite different from the situation where you have the water and graphite in combination.

Mr. **Fazio**. Would it be a calculation that you could agree to, Mr. Salgado, that to retrofit all of your facilities with containment, it would cost somewhere in the neighborhood of $5 billion?

Is that a ball park figure?

Mr. **Salgado**. I think that would probably be somewhere in the low end of the spectrum, and I think there may be a physical problem or a design problem of trying to encompass the entire N-Reactor.

REPLACEMENT FACILITIES

Mr. **Fazio**. You would probably be looking to new facilities if it were decided we would need a safer kind of reactor condition.

What would a new N-Reactor cost for example?

Mr. SALGADO. I think the figures for a new production reactor, I believe, are somewhere in the neighborhood of approximately $2.3 billion to $4.6 billion for one new production reactor.

Mr. FAZIO. Could we, in effect, build one that might provide the total requirements of our existing facilities for that kind of number?

Mr. SALGADO. No.

Mr. FAZIO. What kind of new facilities would we be required to replace the existing production facilities at DOE?

Mr. SALGADO. I do not have that in front of me, and I am not quite sure.

Part of it is an unknown factor, the national security needs of the country and the stockpile. So that varies.

I don't know how many reactors it would be. I can attempt to speculate for the record, if you would so desire.

I think the other salient point is that I believe that our indications are that it would probably take anywhere between probably ten to 12 years for not only the design and the environmental assessments, but to build and have on line a new production reactor.

Mr. FAZIO. Unless we were willing to be without that production capability, we would have to wait ten or 12 years to build a new facility.

We would have to get started almost immediately in order to make the kind of progress that would reassure you that you could meet your national security requirements.

Mr. SALGADO. If, in fact, there was a proven need for a new production reactor and it was determined it was needed in such and such a time frame, you have that time frame or envelope of ten to 12 years, I have been told.

Mr. FAZIO. How much longer do you think the Hanford facility will be productive?

Mr. SALGADO. Sir, that is an ongoing review.

As you have heard earlier, we anticipate by the early 1990's we have to make some determinations as to additional modifications to that facility.

Mr. FAZIO. You are considering modifications, not replacements.

Mr. SALGADO. We are considering modifications at this time, and given what we know, modifications seem to be the prudent path to take.

It is a system that works safely and soundly and has met our needs for over 23 years.

Mr. FAZIO. From what you know now, you wouldn't put containment on a new facility.

You have indicated in your response to Mrs. Boggs you don't have enough evidence to justify changing the method of safety precautions, redundancy, etc., and the confinement technology, is that correct?

Mr. SALGADO. Sir, that is correct.

There is no indication that our facility is being operated in any way or any manner that is unsafe and dangerous to both our workers and the community to which we reside.

We have no information of that now, and we have our record, and we have an ongoing—I stress it is an ongoing dynamic review—of safety processes and procedures.

Naturally the Russian incident has heightened that awareness, and we are continuing to review that in light of the potential from that incident given graphite.

PLUTONIUM PRODUCTION

Mr. **Fazio**. I don't know if you have answered this before. I may have missed it.

Could you tell us, to the extent that you know, what Chernobyl was producing? Was it a plutonium producer? Are they in a defense production mode, or are we talking about commercial energy production? Is it some hybrid, some combination?

Mr. **Salgado**. It is my understanding it is a commercial reactor producing electricity and I have no information to the contrary.

Mr. **Fazio**. All four of those reactors in that area do the same thing?

Mr. **Salgado**. I do not know, sir.

NRC OVERSIGHT

Mr. **Fazio**. Let me just ask one final question.

There are some who think it would be appropriate to have the NRC in a position to oversee and regulate the reactors that are within the purview of the Department of Energy.

Would you comment on your reaction to that suggestion, that proposal?

Mr. **Salgado**. Yes, sir, I would.

I think, first of all, that essentially DOE and NRC regulations are derived from the same national and international standards and guidance.

I think it is important to remember that we regulate the industry, the private industry and the universities that operate our reactors, and we are in a regulating mode.

I think our safety record is excellent and outstanding, that we have clear lines of authority and responsibility to move forward and to address safety needs in a regulatory fashion.

I think it is something that has been—this came out of it historically when the Atomic Energy Commission, I believe, became bifurcated and DOE came into existence, NRC looked to regulation of the commercial reactors and DOE had the responsibility for regulating the safety of the production reactors and government-owned reactors.

Essentially, we see no need to move into another regulatory system. It is important to remember DOE, through its labs and its ongoing expertise and institutional knowledge, has some of the foremost experts throughout the world in the operation of nuclear plants and nuclear safety.

As a matter of fact, I believe it is mandated by law, thus some of NRC's activities, when going outside, come to our national labs to gain that expertise.

I think also that we have a responsibility at DOE not only to insure that we meet the safety requirements but also we meet the national security needs of this country.

Combining both within one department, I think we are able to merge and integrate those systems and move forward effectively.

And I would go back, sir, and again indicate that based on 30 years at Savannah River and 23 years at Richland, Washington, there is no need either to question DOE's ability to regulate our government-owned, contractor-operated facilities.

Mr. FAZIO. I guess the response the critics would give, is NRC puts safety first, regardless of the ratepayer. The critics would say that you may have a conflict of interest.

You have a particular responsibility, maybe, to meet national security requirements and safety may not rank as high in your list of priorities.

Would you want to respond to that?

Mr. SALGADO. I would respond to it indicating that, number one, they can say those things. I don't think our record, our past history, would even indicate that.

I think the wisdom of this committee and other committees that we are called before to insure that we are meeting our safety requirements and moving forward.

I think the other side of it is that Congress, through its appropriations and authorizations, is also able to give direction to the ongoing safety needs of our facilities, and that in many times in the past, many times probably in the future, that you, Mr. Chairman, in your wisdom, will be appropriating funds, authorizing funds for us to move forward in a safety arena you may deem appropriate more than we in the Executive Branch.

I think that is more easily and quickly done than in the commercial side where there may be penalties assessed.

Essentially, I think we are responsive and have a good safety record.

Mr. FAZIO. Thank you very much.

I have some questions from Mr. AuCoin I would like to place in the record.

Mr. BEVILL. Without objection, so ordered.

We have our good friend and colleague, a member of our House Appropriations Committee, Mr. Norm Dicks.

Be glad to hear from you, Norm.

N-REACTOR FY 1987 BUDGET REQUEST

Mr. DICKS. Thank you, Mr. Chairman.

Again, I appreciate your kindness in allowing me to be here today to ask a few questions.

First of all, I am pleased to hear generally the report and the statement regarding the N-Reactor at Hanford.

Now, there has been some concern in the State of Washington because of the budget request this year. Apparently, several hundred fuel tubes at Hanford's N-Reactor are gradually becoming unacceptable for operation because long use is distorting and making them brittle, according to the 1987 budget request document submitted by the Department of Energy.

The documents state failure to replace the tubes which hold nuclear fuel in the reactor's graphite block may result in significant loss of reactor production.

Budget documents and an environmental impact statement obtained yesterday indicate another problem: swelling and expansion of the graphite block as a result of constant bombardment with radioactivity.

The distorted fuel tubes and swelling graphite constitute major technical problems in the Department of Energy's effort to extend the operating life of the 23-year-old reactor.

This was part of a story in the "Seattle Post Intelligencer". Obviously, I have been a strong supporter of the N-Reactor and the joint energy production and the national defense use of the reactor.

We certainly want to make certain that N-Reactor is operating as safely as possible. I guess in that sense we should be pleased you are asking for $800,000 to deal with these problems. I think it is important for the citizenry in our State and in the northwest region to understand, if this is a serious problem.

Just what is the Department doing to try and correct these problems?

Mr. SALGADO. Sir, this is a problem that we have addressed a little earlier. We are approaching it in the early 1990's.

I have Mr. Hunter here from Richland, Washington, and given the tenor of your question, I would ask for a complete, thorough analysis by him.

Mr. Hunter?

Mr. HUNTER. First of all, let me address the $800,000 question.

This was a budget request to correct some deficiencies in the graphite and shield cooling system, not the reactor process tubes. The graphite shielding cooling system is a tertiary cooling system which is primarily intended to provide some cooling to the graphite moderator block itself.

There are tubes in addition to all of the other tubes that run through the core. There are small graphite cooling channels that run through the core, many hundreds of them, and that is the system which this particular budget request addresses.

The purpose of the budget request, however, is not to deal with the cooling tubes themselves. The purpose of the budget request was to provide some seismic stiffeners to parts of the system which are exterior to the core area.

Mr. DICKS. In reading the statement, either it was inartfully drafted or it points out a very significant problem. The statement was from your budget request.

If there was a seismic disturbance, these tubes could be ruptured and could lead to a serious problem.

That is kind of paraphrasing it in a very kind way.

Could you tell us about that? It says you have potential seismic problems in the region. Why do you need the $800,000 to correct this problem.

Isn't that correct, sir?

Mr. HUNTER. Let me address the graphite cooling system.

First of all, this is what we would term a back-up to a back-up system.

It is not a primary emergency core cooling system for our reactor.

The budget request addresses adding of seismic snubbers to certain portions of the system so that in the event of a seismic disturbance, that system also would be available if needed to provide additional cooling capability.

However, the system as designed is not required for the safe shut down of the reactor.

Mr. DICKS. Would it give you redundancy?

Mr. HUNTER. It gives me a belt and suspenders rather than just a belt.

Mr. DICKS. What about this business of the swelling and expansion of the graphite block as a result of constant bombardment with radioactivity?

Are we doing something to correct that? Is there a need for funds to correct that problem?

Mr. HUNTER. Since the reactor was initially started up in 1963, the graphite moderator has been undergoing dimensional changes.

Because the neutron flux in different parts of the core is different, these changes are proceeding at different rates in different portions of the core.

You have to understand that this is a very long-term kind of process taking years to develop.

We are not talking about rapid changes in graphite dimensions. We are talking about long-term dimensional changes.

And they are rather small changes, but their effect is as the graphite grows and distorts, it also tends to cause a certain amount of bowing in the process tubes.

The process tubes are the tubes which contain our fuel elements. As a result of these long-term processes, we have an extensive program at——

Mr. DICKS. Let me ask you this on that very point.

Doesn't it also transmit cooling water vital to plant safety, those same graphite tubes you are talking about?

Mr. HUNTER. The primary coolant system and the emergency coolant systems flow water through the process tubes.

Those are the tubes, larger diameter tubes, which contain our fuel elements.

Mr. DICKS. So, they do both things.

Mr. HUNTER. I have a reactor core cooling system which is my primary coolant system. That is the primary means of cooling the reactor.

In addition and completely separate to that, I have another cooling system, a small cooling system, which provides some cooling to the graphite moderator itself.

They are completely separate and independent systems.

Mr. DICKS. All right.

A 1983 environmental impact statement warns, N-Reactor's operations might be curtailed because, "radiation induced effects on the graphite moderator, says David Simonson, Director of Production Operations at Hanford for the Department of Energy said yesterday. Parts of the N-Reactor's big radioactive graphite block are swelling, changing shape, and increasing in size. Simonson indicated the Energy Department has decided to find some way to provide more space to accommodate swelling of the graphite rather than replace the block."

Is that accurate?

Mr. HUNTER. That is accurate, yes.

We expect that some time in the early 1990's it will be necessary to make some modifications to the moderator in order to accommodate this long-term swelling that I spoke of.

We also are developing the data to support selective process tube replacements, but this is to accommodate a long-term problem which we expect to become a difficulty for us in the 1990's.

HANFORD SAFETY REVIEWS

Mr. Dicks. As I understand it, you had an investigation on safety that you were conducting.

Now you are going to send out agents, I am told, to take a quick look at the reactor and make sure everything is safe in order to keep it operating, is that correct?

Mr. Salgado. If I could clarify, we have had a schedule for a year for all of our major facilities to undergo technical safety appraisals covering not only operations but also the evacuation plans and all other safety operations at these facilities.

Because of the Soviet incident and because of the nexus between the graphite in the Soviet reactor and our graphite, the Secretary has accelerated the technical safety team schedule.

They arrived on site as of yesterday, Monday, with the Assistant Secretary for Environmental, Safety and Health.

They are particularly looking at the safety aspects of the graphite core to insure that, those safety aspects are in place.

We will continue with our accelerated safety analysis schedule over the entire next month, month and a half, but they will be targeted specifically because of the incidents in the Soviet Union.

Parallel to that, we are going back and doing a review of the concept design of that entire facility.

The last one was done in 1981. It has been referred to as the Crawford Report. I have been told that was strictly a design review that was done strictly on paper and there was no site visit.

We will be commencing site visits to insure that the design is safe and operating in an effective manner.

Last but not least, the Secretary has asked a group of external experts to come together this week to review, to insure that we are looking at all the salient safety aspects and technical safety aspects, as well as to explore the known data on the Soviet incident, and to insure that any potential issues raised by the Soviet incident is also addressed by the Department of Energy as we review our N-Reactor.

Those are the three things.

Mr. Dicks. We are going to look at this swelling, the graphite block that is swelling. Is that part of the review?

Mr. Salgado. That is an ongoing process already, sir, and that is not deemed to be part of the, quote, unquote, "technical", though the concept design group will be bringing that into consideration.

All three groups we anticipate, by the end of June, will have completed various segments of the majority of their studies. That is correct.

Mr. Dicks. My own assessment is in order to satisfy public opinion there is going to have to be some type of independent review, whether it is done by the NRC or some panel of outside experts.

Maybe the panel the Secretary is talking about will suffice, but there are a number of my colleagues from the northwest that are talking about trying to get the NRC to take a look at this.

I know they are going to be there shortly. I just want to say this is a matter of grave concern.

EMPLOYMENT AT HANFORD

I would like to ask one final question.

How many jobs are affected here? If they shut down the N-Reactor, how many people would be affected?

I am told it is several thousand.

Mr. **Salgado**. I have been advised, sir, that N-Reactor employs 2,300 people in Hanford.

Mr. **Dicks**. Rather than throwing all these people out of work, there is nothing on the record at this point that indicates the plant should be shut down while this safety inspection is being conducted is that correct?

Mr. **Salgado**. Sir, that is correct.

We have no indication, no empirical data and no hypothesis to indicate that plant is operating in an unsafe manner.

We are moving forward to re-look at this issue, but there is nothing in 23 years of history of safe operation of that plant to indicate that plant should be shut down.

Mr. **Dicks**. You wouldn't hesitate shutting it down if you found a problem, would you sir?

Mr. **Salgado**. If we found a problem that would endanger the health or safety of our workers or the community, we would shut the plant down.

Mr. **Dicks**. Thank you, Mr. Chairman.

Mrs. **Smith**. Mr. Chairman, might I ask two very brief questions?

Mr. **Bevill**. Chair recognizes Mrs. Smith.

AGRICULTURAL IMPACTS IN THE SOVIET UNION

Mrs. **SMITH**. From what you know now about Chernobyl, would you comment about the possible long-term danger either of grain or meat contamination and also loss of productive capacity in that bread basket area of Russia.

Mr. **SALGADO**. Ma'am, I could only respond that we have insufficient data to make any prognostications or render a reasonable opinion as to the long-term effects both in agriculture or throughout the entire industrial complexes around there.

The underlying caveat is that there is a serious accident. We expect some long-term adverse effects. What they are to the degree that they are, I think would be pure speculation at this time.

So I can't provide you that.

NUCLEAR PLANTS WORLDWIDE

Mrs. **SMITH**. Of these 245 nuclear plants that are around the world, you said some are in Russian satellite countries.

Are any in Central or South America?

Mr. **VAUGHAN**. There are reactor plants in South America, yes.

Mrs. **SMITH**. Would you have any further comment about any possible future action to be sure that these plants that are built around the world are as safe as possible?

Mr. **VAUGHAN**. Well, I think, Mrs. Smith, that once again, to the degree we can expect for the noncommunist countries there is a general international consensus that safety standards to protect the health and safety of the public are factors that are considered by every country that I know of and all the professional and government people that we have had conversations with.

Mrs. **SMITH**. Thank you, Mr. Chairman.

Mr. **BEVILL**. Yes. Chair recognizes Mr. Fazio.

SEISMIC TESTING

Mr. **FAZIO**. I just want to clarify one point on this question of the seismic situation.

You indicated earlier, Mr. Salgado, in your response to me, you felt the plants were all safe for seismic purposes.

What is meant by the budget request that you have made that goes to improve safety for seismic purposes? Does it mean that some testing has been done recently to prove to you that you were not as seismically secure as you thought you were?

You had an opportunity to sell us on your budget submission. You didn't take it when Mr. Dicks zeroed in?

Mr. **SALGADO**. If I understand the question correctly, number one.

Mr. **FAZIO**. It is partially a question and partially an assertion, but go ahead.

Mr. **SALGADO**. Let me deal with the question and not the assertion, then, sir.

I would indicate, sir, the budget request went in for a redundant safety system as I understand it. We are talking seismic, and I may not have been explicit, we are talking about that area of the plant where the core is located and those crucial features of the plant that if, in fact, there was a major seismic disturbance, would create a major problem for the core.

It is my understanding that the request that went in the budget was for a redundant system for the cooling of the graphite moderator and is not located within the core where we are satisfied of our seismic stability.

Is that correct?

Mr. HUNTER. That is essentially correct.

Mr. FAZIO. Do you want to comment any further, Mr. Hunter? You said, "essentially correct".

Mr. HUNTER. The graphite cooling system is what we call a tertiary cooling system. In the event of a reactor casualty, it does not provide primary coolant to the core.

You should understand that the solutions we find to deal with seismic events is a whole area that has been evolving since this reactor was built.

If you look back on the history of N-Reactor you will find we have made a number of seismic upgrades to make our systems meet the current requirements that have been found to be appropriate for the industry.

In other words, we are trying our best to keep this reactor up-to-date and that is the reason that this particular item is in the budget; however, it is an enhancement.

It is not an essential upgrade that would take priority over some certain other upgrades that we need to take care of.

Mr. FAZIO. Are there other requests in this budget for seismic upgrades on other reactors or are there several others possible budget requests in the pipeline that will be coming up to us over the next year or two?

Mr. HUNTER. You may find in future budget submissions further requests for seismic upgrades.

Mr. FAZIO. All of these are simply reacting to the newest technology, and newest testing?

Mr. HUNTER. As we learn more we develop and propose more.

Mr. FAZIO. You have no facilities you consider to have a seismic problem?

Mr. SALGADO. When you say facilities, I would imagine we must have hundreds of facilities.

Mr. FAZIO. Reactor facilities.

Mr. SALGADO. Production reactors. The technical people have told us that the core and main structure are meeting seismic demands and taking care of seismic contingencies.

As to outbuildings and pumps and things like that that are not within that core structure, I am not sure whether or not they are meeting various seismic requirements.

Mr. FAZIO. Pump structure could at some point be very important to the operation of the reactor. Are you giving additional attention to these facilities, as well? Because the point you are trying to make seems to qualify the answer that was made.

Mr. SALGADO. Let me ask Savannah River.

Mr. SIRES. Yes, sir.

The seismic analysis includes all support equipment, all those things that are essential to safety, all the safety features as well as the reactor core and the reactor building. The confinement system is also included, all of those.

Mr. **Fazio**. So, anything that is essential to the operation of the reactor you are willing to say at this time is not seismically stressed under any of the new data you have collected?

Mr. **Salgado**. That is correct.

Mr. **Fazio**. I would appreciate some documentation for the record.

Thank you very much.

[The information follows:]

Seismic Analysis

All the production reactors and their essential support facilities have been designed, analyzed, and upgraded to withstand Design Basis Earthquakes for their respective sites. Throughout the life of these facilities, as evolving seismic design requirements or new analysis techniques became known, seismic reanalysis and, if necessary, modifications of these reactors were performed to improve and assure their safe operation.

Mrs. **Boggs**. Mr. Chairman?

Mr. **Bevill**. The Chair recognizes Mrs. Boggs.

POSSIBLE ACCIDENT SCENARIOS

Mrs. **Boggs**. What are the worst possible accidents that could occur at Hanford and Savannah and what would the consequences of those accidents be?

Mr. **Salgado**. Let me ask Mr. Hunter to respond to Hanford and then Mr. Sires.

Mr. **Hunter**. The design basis accident is postulated as the failure of a single process tube which would cause a corresponding melting of the fuel.

There would be—after the safety systems actuate and go through their sequences there would be some environmental release, but it would be well within the established guidelines for that release.

In other words, the reactor structures operate to safely contain that accident.

In addition to the design basis accident which is fully documented in our safety analysis report, other accidents, hypothetical situations, were considered.

The hypothetical accident, one which is considered incredible, is the total loss of all normal and emergency coolant.

In this event, it is hypothesized that up to one-third of the core might be damaged; however, the consequent release of fissionable materials to the environment are, again, within the guidelines established by the Federal regulations.

But so far as we know at this time, that is the worst outcome that we could have.

Our contractor is looking at other scenarios, but we have not yet been able to postulate anything worse than the one I just mentioned.

Mrs. **Boggs**. Thank you. Now please discuss Savannah River.

Mr. **Sires**. Mrs. Boggs, I would like to comment that the Savannah River reactors are low temperature and low pressure, very low; the temperature in the order of 240 degrees Fahrenheit, and the pressure five pounds per square inch.

Because of this, the worst accident, most severe accident that we could postulate, is what we call a reloading error accident.

That would result in three percent core damage. Three percent of the core would be damaged, and there would be some release of radioactivity.

The probability of this happening is about once in a million years.

The consequences of this to the population would be extremely low.

The maximum exposures off-site would be on the order of less than one rem of exposure, either whole body radiation or exposure to the thyroid.

This is the basis upon which we assess the total system, reactor system, including the confinement.

I might point out that the majority of any radioactive material that would be released in an incident like this would be contained in the confinement system.

It has a series of filters that remove the particulate material as well as the halogens from the gas stream leaving the reactor.

Mrs. **Boggs**. Do you think that the people who operated Chernobyl believed they could not have an accident of such consequences?

Mr. **Sires**. Do I think so?

Mrs. **Boggs**. Yes.

Mr. **Sires**. That is speculation. It is hard to say. I really don't have a basis to speculate on that.

Mrs. **Boggs**. So you think a burning graphite core like Chernobyl's is not possible in an N-Reactor?

Mr. **Sires**. It is not possible in the Savannah River reactor, and that is the one I am speaking to.

Mr. Hunter should speak to N-Reactor.

Mr. **Hunter**. I think a disaster like what happened at Chernobyl is virtually impossible at N-Reactor.

Mrs. **Boggs**. Thank you, Mr. Chairman.

Mr. **Myers**. Mr. Chairman, I have one comment or statement.

Mr. **Bevill**. All right.

REPORTING REQUIREMENTS

Mr. **Myers**. It is obvious the Soviet Union felt no responsibility to alert the rest of the world that there was a potential hazard here in the accident at Chernobyl.

I hope that you will use your office—and I realize this is primarily the State Department's responsibility—but that we put some teeth into the International Atomic Energy Agency. There should be a requirement that any time in the future—we hope and pray it will never happen again—but should there be such an incident, it be reported to the world so the world could be on alert.

I am sure eastern Europe would have appreciated it this time.

I hope we will try to do everything we can to make sure these kinds of incidents can't happen again in the future.

Thank each of you for your testimony this morning.

Mr. **Bevill**. We appreciate the testimony you presented.

I noticed one thing that I should made clear when you are talking about your request for funding.

The requests were made for that funding many months ago and have nothing to do with this accident in Russia. I thought that ought to be pointed out.

We do appreciate your testimony and what you have presented here to us and what you are doing.

We will ask you now to exchange seats with the Nuclear Regulatory Commission.

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TUESDAY, MAY 6, 1986.

U.S. NUCLEAR REGULATORY COMMISSION

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HAROLD DENTON, DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION

Mr. BEVILL. Chairman Palladino, we will be glad to hear from you. We are delighted to have you and your colleagues with us today. You may proceed as you wish.

Mr. PALLADINO. Thank you, Mr. Chairman.

At your request, the Commission appears before the subcommittee today to discuss the potential impact of the Soviet accident at Chernobyl on U.S. Nuclear Regulatory Commission safety programs.

With me are my fellow Commissioners, and I have also asked Mr. Denton to participate. We also have other support staff here.

Mr. Roberts was not able to be with us today. He had to attend a funeral.

As you know, details regarding the accident are sketchy. However, the available evidence indicates that the accident was serious. My fellow Commissioners and I would like to express our condolences to those Soviet citizens, and others, who have been or might be impacted because of the accident at Chernobyl. It is obviously a matter of deep concern to us all when tragedies occur. The President has offered to provide appropriate medical and technical assistance.

The White House has established an interagency task force to monitor the health, safety and environmental consequences of the Chernobyl accident. The task force is chaired by Lee Thomas, Administrator of the U.S. Environmental Protection Agency. Members represent various Federal agencies, including the Nuclear Regulatory Commission. The NRC representative is Mr. Harold Denton, Director, Office of Nuclear Reactor Regulation.

An NRC Incident Tracking Team was established on May 1, 1986 to collect information and review the effects of the Chernobyl incident in support of EPA. The purpose of the effort is to obtain an understanding of the reactor accident and the radiological source term in order to assist EPA in assessing its input to the United States.

As you are aware, Mr. Chairman, we had a serious accident in the United States at the Three Mile Island Unit 2 Facility. Our study of that accident produced an extensive list of changes that the Commission concluded were necessary to improve the safety of nuclear plants in the United States.

Because of the significant differences between the commercial nuclear plants in operation in the United States and the Chernobyl nuclear facility in the Soviet Union, it is difficult to anticipate any lessons to be learned from this accident that might be applicable to plants we regulate.

We, of course, will make every effort to learn what we can from the Soviets, but until we have sufficient information, it is premature to speculate whether any changes in United States commercial plants are warranted.

I have asked the Executive Director for Operations (EDO) to appoint a group of our senior scientists and engineers to continue the study of the accident and recommend to the Commission any action that might be needed for the U.S. regulatory program.

I would like next to turn to Mr. Denton, who will provide an opening statement describing our existing knowledge of the Soviet accident and how it may relate to U.S. nuclear reactor safety. We will then be glad to respond to any questions you may have.

Mr. **Bevill**. Mr. Denton, we are looking forward to hearing from you and we are delighted you are on the President's task force. We look forward to hearing your testimony.

Mr. **Denton**. Thank you, Mr. Chairman.

This accident came to our attention as a result of measurements that were detected in Finland, Sweden, Norway about a week ago.

Based on these measurements and back calculations to Chernobyl, we determined it had experienced a very severe accident releasing tens to hundreds of millions of radioactive fission products into the air.

These were extrapolations of our knowledge of the situation that existed. Soviets have confirmed in today's paper the existence of a fire and an explosion at that facility.

We think a large amount of the fission products were ejected into the upper atmosphere. They have also indicated that the doses around the plant are such as would be lethal if people were to remain in the vicinity for any period of time. They appear to have evacuated out to a distance of at least 30 kilometers and several towns have been evacuated.

Prior to the accident, we have never had any safety interchange with the Russians in over eight years, so we had no current information on the design of that plant. Based on a crash effort to pull out from the literature and from other sources design information, we have learned that this is the fourth unit at that site.

These two units called three and four are different than the first two units; they have what I call a containment of sorts at units three and four. Units one and two did not have a containment of any kind.

The design of the reactor is quite different than commercial U.S. nuclear plants. It is graphite moderated, as you have heard. The graphite moderator can be thought of as a large block of graphite, essentially almost 2,000 tons of graphite, penetrated by 1,600 pressure tubes. These pressure tubes are made of zirconium alloy and house the zirconium-alloy-clad fuel. This design has a much greater amount of zirconium in the core than U.S. designs.

There is no information to tell us what really initiated the event. It could have been a loss of coolant due to a pipe break, a refueling accident, a loss of AC power, or human error. We have no knowledge of the initiating event.

Obviously, from the amount of damage that has occurred, there was what we could conventionally call a meltdown in the sense that the fuel overheated and released fission products. This led to production of combustible gases either in the core or above the core.

The chemical explosion that resulted when these burned blew the roof off the top of the building resulting in the release of these fission products directly to the atmosphere so there is widespread contamination close in.

As a result of putting this into the atmosphere, it drifted around and was deposited on a number of countries in Europe. We have had reports in from most of the European countries of detection of fission products, including Italy.

It was detected in Japan yesterday. Rain water was reported to have concentrations in the order of 10,000 picocuries per liter. Radiation was being found in food stuffs, cabbage, lettuce, these sort of leafy vegetables.

There is every indication that the higher altitude fission products are already passed over the U.S. We would expect that high altitude rain could well bring some of these down to the U.S. today, for example.

As of last night, none has actually been detected at the ground level in the U.S., but we expect it will be only a very short time before the lower altitude levels reach the U.S. and begin to be detected on a rather widespread basis.

Just to repeat what Lee Thomas, the Administrator of the Task Force, has said, there is no reason to believe these will result in a health hazard to the U.S. public.

Mr. BEVILL. I understand that the design of the Russian reactor is unique. Would you compare its design to the typical United States civilian nuclear plant?

Mr. DENTON. The typical U.S. plant, the one we talked about after TMI, is a light water plant. Fuel is roughly 200 tons of uranium and it is surrounded by water and it is enclosed in a massive reactive pressure vessel which is further enclosed in what we call containment buildings.

It has been the practice of the NRC to require containment buildings since its inception. You may recall Shippingport which was built by Admiral Rickover, had the first containment building. So in these U.S. plants of the light water design, there is no graphite to burn.

I think one lesson I draw from this comparison is that in U.S. water designs there is a capability to recover from very severe accidents. At TMI, the core was severely damaged, but in spite of that, we were able to get water back in the core and prevent the release and widespread contamination of land around TMI.

It appears that in the Russian reactor design, once it reached a certain stage they were not able to effectively contain it and it did result in widespread release. So I think between the light water design and the graphite design, there may not be a lot of lessons to draw.

One lesson that came out of TMI was that we needed to design the containments for the production of hydrogen. We recognized at TMI there was a possibility of combustible gases. We reviewed all our containments and we have added now capability to withstand the production of hydrogen in the light water reactor containments.

There is only one plant that we license that has a graphite moderator. That is Fort St. Vrain. It is quite a bit different than the Russian design.

To begin with, it does not have fuel rods, it has micro pellets. The fuel is imbedded in the graphite pellets so there are no zirconium rods.

Secondly, it does not use water to cool the core. It uses helium as a coolant, which is an inert gas, so that water does not interact with the fuel or with the graphite. It does have a very large, high strength concrete pressure vessel. Right after I learned of the Russian events, I did ask the staff and the utility to do a review of that plant on two questions.

One was what was the likelihood that a graphite fire could occur in spite of all the precautions taken to prevent it and, secondly, could they cope with it if it were to occur. That is being studied and it looks like a graphite fire there would not have the opportunity to get air into the plant nearly the way the Russian reactor did.

Of course, another major difference that was mentioned earlier, the Russian reactor is fueled on line so there is a need to continuously, in fact, have open access to the core in the Russian design.

All the commercial U.S. reactors shut down for refueling. That is sort of a brief summary. We expect to come to a recommendation to the Commission for St. Vrain in the very near future.

Mr. **Bevill**. Would you give us what factual information you have as to the radiation that has escaped to countries other than the Soviet Union?

Mr. **Denton**. There is a great deal of information which has been made available by all the European countries and there was a meeting Friday of the members of the Nuclear Energy Agency to share that data.

I think the levels do not rise in general to the health levels for which the health authorities are requiring action. The radiation seems higher in some localities than in others. In other words, it depends on the local meteorology and the rain out. Because of this and the widespread contamination in some spots, there has been some suggestion to not drink the milk or to be sure to wash the vegetables and these sorts of things, but that remains to be determined by the health authorities of each country.

In general, the radiation levels in Sweden and Poland were quite high and because of the decay that takes place each day the activity levels in the other countries are lower.

Mr. **Bevill**. From the information you have, do you know of any reason that we should feel that our reactors in this country are unsafe?

Mr. **Denton**. Based on the information I have, I don't feel that our reactors are unsafe, but I do want to make the point that we have very little information about the real cause of this event. It may be weeks and months before the Russians share with us the exact cause.

I would like the opportunity to re-examine that question when the facts are in. But we are following it very closely. We have a team of people working day-to-day on this problem and because of the differences in design, it may not be directly applicable to the U.S. light water design, although we are very interested in the amounts of radioactivity that did get out and how they were dispersed and it may have some bearing on our research that we have done on source terms.

Mr. **Bevill**. Is it expected that the countries throughout the world that have nuclear reactors report an accident to the International Atomic Energy Agency?

So far as you know, has the Soviet Union ever reported to the IAEA?

Mr. **Denton**. We have talked about that on the task force and with the Department of State. Russia is a member of the IAEA. They have reported about 50 events to the IAEA, but as I understand it, they do not have a legally binding obligation to report.

It is more of a voluntary reporting system.

Mr. **Bevill**. Did they report this one?

Mr. **Denton**. They did not report it for many days.

Mr. **Bevill**. The Chair recognizes Mr. Myers.

Mr. **Myers**. Thank you, Mr. Chairman.

We are concerned, of course, about safety and members of the Nuclear Regulatory Commission have appeared many times before this Subcommittee. The record of this country is one that we certainly should be proud but we cannot rest on our laurels either. That is the reason the Committee felt we should have this exchange today to find out if there is something we should be doing in light of this accident.

It appears the Russians did have some notice that they had a problem at Chernobyl, but they could not control it or did not. You say it is impossible to speculate just what happened but they at least should have known something was wrong. We must know that regardless of our limited knowledge about the reactor that they operate.

It just didn't happen instantaneously. They would have had some advance notice that they had a problem, would they not?

Mr. **Denton**. I think we think they did have some advance notice. It may have been only hours, but the experience almost worldwide with reactors is you don't just have a sudden problem develop. It usually is a combination of human errors and equipment malfunctions and they turn into bigger problems.

So we would just speculate they may have had a small problem which got away from them and perhaps they were trying to cope with it and it was continually degrading.

Mr. **Myers**. It appears from previous testimony that if we had not had the containment facility at TMI, the accident there would have been more serious than it was. Did we exchange with the Russians at that time, any information about our experience at TMI? The design of the two reactors is quite different.

In the case of a hazard like this or accident like this, recovery procedure would be somewhat similar, I would think. Did we share our information with the Russians? You mentioned earlier, Chairman Palladino, we had not had a safety exchange with the Soviets for the past eight years.

Did we offer any advice on what we experienced at TMI?

Mr. PALLADINO. I didn't make the statement, but nevertheless, we made all our information available to the public. I am not sure that we overtly sent it to the Russians, but I am confident the Russians were looking at all the information we had.

Mr. DENTON. We made it available worldwide and there are indications that the Russians did make some changes after TMI.

In fact, unit four ironically is the most recent unit on this site. It does have compartments of containment. It is not fully contained the way we think about it, but it is clear that they did move toward containing certain types of events.

But they didn't contain the entire plant in the type of large vessel that we think of.

Very clearly, though, they did put in in this type of plant an ability to withstand some types of reactor failures.

Mr. MYERS. Unit number four at Chernobyl had the accident and went down; is that correct?

Was it brought on line after the TMI incident?

Mr. DENTON. Yes, sir. Came on line in 1983.

Mr. MYERS. That was four years later, so they had every opportunity to gain from our experience.

Mr. DENTON. In looking at the literature available, they do have a type of emergency core cooling system, so I don't want to leave the committee with an attitude they did not recognize safety totally. I don't know how effective their safety reviews were, but you do find attention being given to many of the same kinds of safety issues that developed here after TMI.

Mr. MYERS. Under our procedure, when TMI had their incident, did we notify the rest of the world under the agreement we have with the International Atomic Energy Agency?

Mr. DENTON. Yes, we did. We made a major attempt to keep them and all other agencies fully informed. From a factual basis, in fairness, it was several months before we produced reports that began to detail exactly what happened, but we sure kept the world informed on what was happening the moment we knew it.

Mr. MYERS. We could have had an explosion there, hydrogen explosion?

Mr. DENTON. In fact, we did have a hydrogen explosion on Thursday and it was about 28 pounds per square inch and did not threaten the integrity of the containment which was designed for more like 50 pounds.

Since TMI, we have destructively tested models of containments and find they can really withstand two or three times the amount of pressure that they are designed for. It appears that the use of a containment building or really good containment system can reduce off-site consequences by many orders of magnitude, since the Russians were not able to confine theirs.

Mr. Myers. Mr. Asselstine.

Mr. Asselstine. I was going to underscore this containment/confinement. There has been a lot of talk about the early Soviet plants not having containments or confinements. We received some information last night on this containment/confinement system that this particular Soviet plant had, the one that had the accident. It does appear to be similar in many respects to the kind of containment system that the G.E. boiling water reactors have in this country with use of special pools and even with higher pressure levels of 27 PSI in certain portions and 57 pounds pressure in other portions.

So to a certain extent I agree with Mr. Denton. It appears after TMI the Soviets did make some improvements in the containment area in ways that appear to be very similar to the boiling water reactors in this country, but it does not appear that containment structure worked in this case.

Mr. Myers. Do we have any evidence that the other reactors at Chernobyl might have had damage other than reactor number four?

Mr. Denton. Let me just put on an addendum to what Commissioner Asselstine said.

We have looked at that containment and it does incorporate some of the suppression pressure features that G.E. has, but it is markedly different in the region of the core. They do seem to have over in some parts of the building a capability that looks somewhat like G.E. but I don't think it should be considered to be the kind of containment that G.E. puts in their plants.

They typically have a vessel that contains the reactor vessel and the piping and everything else. This Russian design is a compartmental design and is not a thorough comparison.

Mr. Myers. From the evidence——

Mr. Denton. Based on the evidence we had as of yesterday, there was no evidence that unit three was involved, but we have been concerned all along about three because the systems are interconnected to some extent.

The control room is shared so in the first early days we were worried that perhaps due to the limited access to unit three that unit three might become involved. Since unit three has not gotten involved to date, the power level in unit three is down considerably to maybe two-tenths percent of what it was at full power and the amount of water it takes now to cool unit three is probably less than 100 gallons a minute.

We think in all likelihood the Russians were able to get that amount of cooling in the unit pretty soon. There is no indication unit three is involved. Units one and two are physically separate and there is no indication they may be involved.

I think activities are going on in units one and two.

Mr. Myers. From the evidence we have seen from Chernobyl, the appearance of the buildings look similar to those at Hanford. The roof went off the building at Chernobyl and some of the concrete structure around it shows it had problems. It worries me when we have the similarities.

We have vessels at Hanford that would probably protect it. You described the other day to the country that Chernobyl was not a melt down, that it was a burn down. Could you further amplify what the difference is and is that significant?

Mr. **Denton**. Not much difference in what has been released to the atmosphere and the environment. What I was trying to distinguish between the jargon of a meltdown in the U.S. where it is the reactor core slumping where the water lands in the bottom and cools there or melts down to the bottom of the reactor vessel.

This case in Chernobyl where the fuel is so dispersed through this large graphite block I think the fuel reached temperatures where it would melt and at the same time or about the same time the graphite reached temperatures where it would burn.

So, in effect, it was fuel and graphite burning and instead of a molten glob of just uranium fuel, it was a widespread fire that sort of consumed the core, but the net product was the same, namely, release of the fission products that were contained in the fuel. I don't see this one having a glob of fuel, I just see a large charcoal block that has been destroyed.

Mr. **Myers**. It is obvious there was an explosion. The fuel that both the Russians and the United States use in their commercial reactors is low enriched fuel that can't explode in its present configuration. However, with the chemical reaction from the fire and the graphite, is there a possibility that distorted the characteristics of the fuel and made it explosive?

Mr. **Denton**. I think it was a chemical explosion, Mr. Myers. The interactions between steams or commingled graphite can easily produce an explosion that would have the effect that we have seen at the plant.

Mr. **Myers**. Then the explosion just merely removed the integrity of the building. It didn't change the nuclear reaction, but it removed the protection provided by the building.

Mr. **Denton**. That is correct. And the data we have indicates that the building was just a light industrial type roof over the refueling floor of the plant and it could not withstand the kind of explosion that occurred at TMI, for example.

Mr. **Myers**. I have some other questions, but in the interest of time, I will put them in the record. Thank you very much for your testimony.

Mr. **Bevill**. The Chair recognizes Mrs. Boggs.

Mrs. **Boggs**. Thank you, Mr. Chairman.

Thank all of you for being here.

In the other countries who belong to the international organization, who evaluates accidents and how is that evaluation coordinated and released to the rest of the world?

Mr. **Denton**. Are you discussing the Chernobyl accident specifically or just in general?

Mrs. **Boggs**. Just in general.

Mr. **Denton**. In general, there is very good cooperation among most countries of the world. In other words, for the member countries of the Nuclear Energy Agency which include France, Germany, Spain, Italy, Japan, Canada, I would say there is excellent coordination.

Russia is not a member of that agency. But the practices among those countries are very uniform and, in fact, many countries have adopted the NRC's regulation as sort of the base from which they have developed their safety criteria.

Mr. **Palladino**. They have also used many U.S. codes for calculation purposes and for analysis.

Mrs. **Boggs**. So the information we are receiving from other countries, can be evaluated easily.

Mr. **Palladino**. At least those countries that are in the NEA.

Mrs. **Boggs**. What type of emergency responses are there for the population around the Hanford and Savannah River reactor sites? How do they differ, if they do, from those around commercial sites?

Mr. **Palladino**. I am not familiar with their emergency response procedures. I don't know if anyone here is.

Mr. **Asselstine**. Mrs. Boggs, since we don't license or regulate the production reactors, that is a question that DOE probably would have to answer.

I am not sure we are familiar enough with what they do to be able to give an answer.

Mr. **Zech**. I think it is more appropriately asked of the DOE officials, Mrs. Boggs.

Mrs. **Boggs**. Since they have left, I will have to ask it for the record.

Mr. **Fazio**. They are still here.

Mr. **Salgado**. We could provide the detailed analysis of our emergency evacuation plans. They are extensive and well coordinated throughout the Federal system. In the Hanford facility and the ten mile radius around the plant, DOE has contacted all home owners to advise them what the emergency plans are and to take further steps to assure all people within a ten mile radius of that facility are aware of the emergency plan.

I would ask that we submit for the record some more detailed analysis of our emergency plans and evacuation plans both for Hanford and Savannah River.

[The information follows:]

EMERGENCY RESPONSE PLANS

The Savannah River Plant (SRP) maintains a comprehensive set of emergency preparedness plans and procedures to support both onsite and offsite emergency response actions in the event of an accident or incident. The plans and procedures are an integral part of SRP's emergency management system which includes activation of critical emergency management facilities, effective deployment of resources (manpower and equipment), accident mitigation, notification and communications, protective actions for employees and the public and technical exchange of information with State/local officials.

The level of preparedness at SRP is consistent with technical descriptions of potential incidents/accidents as defined by approved Safety Analysis Reports. The geographical boundaries of the site (300 square miles/192,000 acres) provide a substantial buffer zone between major operating areas and population centers offsite. Conditions for activation of emergency response resources are contained within the plans and are consistent in approach as applied by DOE and its contractors. SRP maintains a 24-hour emergency management facility which is staffed by trained personnel to insure a rapid response to emergency occurrences. This facility is in constant communication with key decision makers in the event emergency response actions are required.

Memoranda of Understanding (MOU) which specify offsite notification requirements are provided and maintained current between DOE and the States of South Carolina and Georgia. All plans and procedures which establish emergency management interaction with offsite officials are consistent with the guidance and requirements provided by the respective State and local officials and organizations having jurisdiction over these matters.

Offsite emergency planning zones have been designated and specified in plans and procedures. State and local site specific emergency response plans have been developed for the SRP site.

SRP maintains an aggressive emergency response training program. On average, eight emergency management drills are conducted annually. State and local participation has been included three times over the past 15 months. Tests to date have indicated that the program in place is sound.

All plans and procedures are compatible with the planning and guidance provided by DOE, NRC and FEMA, in order to assure compatibility with the emergency plans of a commercial power reactor located near the Savannah River site.

The Richland Operations Office (RL), Department of Energy, maintains a comprehensive set of emergency preparedness plans and procedures to support both onsite and offsite emergency response actions in the event of an accident or incident. These plans and procedures are totally integrated with the onsite contractor, States of Washingtion and Oregon and local emergency plans, procedures and response actions.

RL's emergency management system includes activation of emergency management facilities, effective deployment of equipment and manpower, notification and communications, protective measures for employees and the public and accident coordination with State and local officials and other Federal agencies.

Accident analyses have been completed for all nuclear facilities at the Hanford Site and the level of prepardness is based upon the accidents listed in the Safety Analysis Reports for those facilities.

Conditions for activation of emergency response resources are contained within the plans and are consistent in approach as applied by DOE, its contractors, State and local governments. RL maintains a 24-hour emergency operation center staffed by trained personnel to insure rapid response to emergency occurrences. RL staff and management duty officers are on call and available on a 24-hour basis.

Memoranda of Understanding are in place with Federal, State and Local governments, as well as all other agencies in which RL coordinates its response actions.

All plans and procedures are compatible with the planning guidance provided by DOE, NRC, and FEMA, in order to assure compatibility with the emergency plans of a commercial power reactor located on the Hanford Site.

Offsite emergency planning zones have been established and people living within these planning zones have been personnally contacted and provided with documented emergency instructions. Notification systems are in place to notity the public of an emergency.

In summary, both the Richland and Savannah River sites of the Department of Energy (DOE) maintain comprehensive sets of emergency preparedness plans and procedures to support onsite and offsite emergency management actions in the event of an accident. These plans and procedures are totally integrated with those of offsite authorities. They reflect conservative estimates of the consequences of the most severe credible accident at each nuclear facility on these sites. A major exercise is conducted annually at both Hanford and Savannah River, frequent drills are conducted at each facility, and all personnel are required to participate in the emergency preparedness training programs. In addition, emergency preparedness is included as a topic in the extensive DOE site and facility safety appraisal program. High priority is given to emergency preparedness in the budget process to assure that adequate resources are available for these activites.

Mrs. **Boggs**. We have an interagency task force working lead by Mr. Thomas of EPA. Since EPA and FEMA both fall under responsibilities that I have on another subcommittee, I feel that it is very important that we have this interagency task force. It disturbs me that DOE is not familiar with NRC's evacuation and safety plans nor is the NRC aware of DOE's.

I do think that we should indeed coordinate our information and our efforts. For instance, the commercial plants take into account that there are containment buildings. Given the lack of containment buildings, how would the time available for evacuation or other emergency response measures compare to NRC licensed reactors?

Mr. **Palladino**. Well, it would depend on the character of the accident that we are discussing. Even for accidents involving the commercial reactors, there would be different response times available for different kinds of accidents. But as for the difference between the two, I would expect that the amount of time for comparable accidents wouldn't be different, whether it is a commercial reactor or N-Reactor or one of the others.

Mr. **Asselstine**. Mrs. Boggs, I wonder if I might add something for purposes of completeness in terms of what our involvement has been with the production reactors. Before the NRC was set up, I believe that the regulatory staff and the Advisory Committee on Reactor Safeguards did do safety reviews of the production reactors—I think in many instances as they were planned and developed. Those reviews, according to the staff, identified a number of safety concerns about the production reactors including questions about the confinement. I doubt they went to the question of emergency planning, but they did go to questions of accident risks for the plants without confinements, the absence of confinements.

All of those reviews stopped in 1974 when the NRC was established, but there were a series of reviews prior to 1974 that did identify some safety concerns about the production reactors. Now, I had been aware of the reviews. But we didn't realize or I didn't realize until last night when we received the briefing materials for this hearing that those reviews had identified a number of safety concerns about the production reactors including going to questions about this confinement concept, seismic adequacies for the production reactors.

I haven't had a chance yet to look at those letters. They may have been corrected by subsequent safety reviews by DOE, but I think we just don't know at this point. All we know is there were concerns that were identified based upon the differences between the designs of those plants and the commercial plants.

Mrs. **Boggs**. Thank you. What does the Commission believe the likelihood of a severe core melt accident to be in the next 20 years for the licensed reactors now operating and those expected to operate?

Mr. **Palladino**. We have made a number of calculations. Among them, one that has been quoted widely has been that if we assume that every reactor had a core melt probability of three times ten to the minus four or three chances in ten thousand then with a hundred reactors operating, and assuming distributions and assuming there has been no changes in design, this would result in a 45 percent probability that an accident similar to TMI could take place.

But there have been changes going on in the plant. Not all the plants are three times ten to the minus four. Many are better and I am not sure if any are worse, but I will turn to my colleagues on the staff for that.

Changes have been made, such as dealing with the hydrogen question, that tend to impact on the number, but nevertheless, if one makes that calculation that is the number you get.

Mrs. **Boggs**. Does anyone on the staff want to supplement that information?

Mr. **Zech**. I think it is important to say that when you talk about that number, you are talking about a core melt and you are not talking about operator actions and other things. It is a very academic number to arrive at.

On the other hand, it also does not mean that there is any radiation release to the public at all because the containment will accommodate a great deal of that. So I think when you talk about any significant release to the public of radiation, you get down to something of a number like one-half of one percent.

But when you are talking simply core melt is when you get to the higher percentage number.

Mr. **Denton**. I just want to make the point we have some very good performing plants in the U.S. We are trying to get more into that category.

If you take the plants around the Great Lakes, for example, something about their management, their design, those plants run as well as any in the U.S.

So does Yankee which is 25 years old and that plant runs exceedingly well. We did have some problems last year with Davis Besse, you recall, and some other plants. So I think the focus tends to be to bring up the lower 20 percent in the U.S. so that we change this probability by working on the poor performers.

Often you will find in looking at the data it is not the design of the plant, but it is the competence, the effectiveness, the dedication of the management. If there is one area I think the Commission is trying to improve, it is in that area.

Mrs. **Boggs**. The gentleman knows this Committee member has been very interested in that phase of the whole operation, of course.

Mr. **Asselstine**. I wonder if I might add a comment on that. I think that is a fairly important number.

There is no question we have had a good safety record in the nuclear power program in this country with the possible exception of TMI and even TMI by the best estimates did not result in serious health consequences certainly of the type that the Soviet Union is experiencing.

But if we have an estimate that shows that there is about a 45 percent chance of a core meltdown accident in the next 20 years in this country, it seems one of the lessons from the Chernobyl accident is for us to decide if we are really satisfied with that level of performance and if we think we can tolerate that kind of accident condition.

It may well be that the next core meltdown accident in this country is arrested just as TMI was and there won't be releases from the containment, but I think we are also learning from much of the severe accident work we are doing that there are major uncertainties about containment performance.

If you get to the position where you have a melted core, there are lots of uncertainties about how the containment will perform, how the core will react with the concrete in the containment or with water in the containment.

There are uncertainties about what condition the containment will be in at the time of the accident. We have had incidents in which containment integrity has been lost in this country from time to time.

So it seems that perhaps one of the lessons of Chernobyl is for us to decide that we are not comfortable with that level of performance or safety and as Mr. Denton says, focus on those that aren't performing up to the level that we would like to see. Then, take some forward-looking initiatives to reduce that probability substantially and to increase the chance that first, we can prevent this kind of an accident from ever happening, or if it does happen, that we have a high degree of confidence that the consequences really will be mitigated.

I think for that reason it is an important number and what that number tells me is that we ought to be doing some more to drive down that risk, even more than we are doing now.

Mr. **PALLADINO**. I wonder if I could add an additional point?

One of the problems we have is in computing core melt. Let me point out that in making these calculations what is done is as soon as the water has gotten below an acceptable level, it is assumed that the core melted.

That is quite a conservative assumption that must not be forgotten here, because there are a lot of things you can do and that have been done in incidents to change the circumstances after you get to that particular level of water flow.

Mr. **ASSELSTINE**. One final point, Mrs. Boggs, if I could.

We have the calculations and that is one thing. As Mr. Denton said, we have some plants in this country that are truly outstanding performers, just as good as any other plants in the world.

At the same time, the operating experience of some other plants is showing us that we still are seeing some widespread failures of equipment and the kinds of combinations of human failures, equipment failures and system malfunctions that can get you into the situation where you face a real likelihood of another severe accident.

I think it is not just the calculations and the numbers, it is also the operating experience at some of the plants that provides a strong basis for our continuing to pursue some forward-looking safety improvement programs.

Mr. **PALLADINO**. I would like to add, though, I don't think this should imply our plants are not safe. As a matter of fact, if any of us thought it was not safe we should immediately take steps to try to get them shut down.

There has been a lot of newspaper talk about 1985 being the worst year. I think we should look carefully at that because in terms of abnormal events, 1985 was not. At least in numbers, it was less than 1984. When you consider we have added plants and you look at the normal occurrences per plant, our situation has pretty well stabilized actually with a slight downward trend.

But I agree with my colleagues that it is not enough to count abnormal events. What I gain my confidence from, among other things, is the fact that we have put in equipment to cope with a wide variety of accidents and that we have developed procedures that permit us to cope with a wide variety of accidents.

We have trained people who cope with accidents in a variety of circumstances and even some of the events that have taken place which we don't like to see. The combination of these factors has permitted the safe shutdown of these reactors under challenging or adverse conditions.

Mr. **Asselstine**. The last point that the Chairman made I think is worth looking at. I don't think that there is a basis for the immediate shutdown of plants in this country. There are a couple of licenses that are shut down now that I think ought to stay shut down until some major improvements are made in the way they operate their plants, but I don't see the basis for an immediate shutdown of the operating plants.

But I think you can distinguish an immediate threat from whether you are really comfortable with the level of performance over the next 30 or 40 year life of some of these plants. In my view, you can say there is not a reason for shutting down plants now, but we are not comfortable with the level of performance such that we are prepared to tolerate that level of performance for 30 or 40 years.

I think we can say we ought to strive for continued improvement where it is cost effective and particularly in the case of some plants we ought to focus attention to bring that level of performance up to an acceptable level so that we are really comfortable for the long term operation of the plants.

Mrs. **Boggs**. Thank you very much.

Mr. **Bevill**. The Chair recognizes Mrs. Smith.

Mrs. **Smith**. Thank you, Mr. Chairman.

Thank you for being here. I have four questions.

Mr. Asselstine, you said you thought more should be done to drive down the risk.

The Administration requests $98.6 million for NRC's inspections and enforcement programs.

Do you think that is adequate, and what would be your comment?

Mr. **Palladino**. My comments or Mr. Asselstine's?

Mrs. **Smith**. Whoever's comments.

Mr. **Palladino**. My feeling is that we are getting very, very marginal in the amount of money this agency is getting, and I think we should watch very closely that we don't get the amounts down to the point where we can't do the job as we see it.

But you must understand that when it comes to inspection activities we only audit. We get the applicant to set up a quality control procedure, quality assurance procedure, a training program for its people, operational procedures, and we audit these various functions.

Compared to the number of people that are involved on the part of the licensee, say in a construction project where it may be several thousands, we have at best one or two persons that spend time inspecting these various activities. We depend on the licensee to do it.

I really think that our inspection budget is probably as low as it ought to go. As a matter of fact, I think it ought to go upward.

I have made statements also, I believe, that we in the Commission ought to be doing more requalifications of reactor operators, because I think that there is merit in the Commission doing it, at least on a more frequent basis so that it keeps the people knowledgable about the plants that they have.

So, you asked me do I think it is adequate. I think it is barely adequate, but I would suggest that just studying an incident such as we have described is going to require some additional funds for the NRC.

Mrs. SMITH. Thank you.

Mr. Bernthal?

Mr. BERNTHAL Mrs. Smith, the Commission took rather large budget reductions in its overall funding level over the last two years.

I guess the principal one actually came as an adjustment in the Fiscal Year 1986 budget.

I think the thing to make clear here is that there doesn't suddenly come a point at which the public health and safety is in danger.

There comes an erosion. It is the erosion that I think needs to evoke some concern on our part.

Chairman Simpson over on the Senate side recently asked a similar question and I replied to him in letter that we ought to bear in mind the kind of erosion, apparently, that occurred in the U.S. space program that led to the Shuttle Challenger disaster.

I don't mean to suggest that there is anything that I am aware of that is comparable to that in our oversight responsibility, but the point is there was a choice made way back when to go with the seamed booster design instead of a seamless design, primarily on the basis of cost.

It is a number of these small factors accumulated over a period of time that can ultimately lead to serious safety concerns.

I won't say that right now we are chewing into the muscle. I am prepared to say and admit there has been some fat in our budget. There probably are other areas in which we can make savings, for example, in administration, but we are getting pretty close to chewing into the muscle.

If that kind of cut that seems to be in prospect for many other agencies perhaps next year should come into play for the NRC, then I simply wouldn't be prepared to sit here and say the public health and safety could be assured of adequate protection.

I think we are just about at the margin now where we shouldn't be nibbling away any further.

Mrs. SMITH. My next question was going to deal with what you spoke of Mr. Bernthal.

I will be going home this weekend, and I know what folks are going to be asking me. They are going to be saying there was a warning in the Space Shuttle program. We should have known that there was danger, and we went ahead.

Now, you said, Mr. Denton, that very seldom is there an accident when there isn't some warning.

I think we should be as sure as human beings can possibly be, that if we have a warning we shut down the plant and don't go on.

Mr. Palladino, you said that if you thought there was real danger we would shut down the plant. But this is a case where we must be very, very sure that we don't err on the side of being careless, as they did.

Do you want to make a comment on that, too, Mr. Asselstine?

Mr. ASSELSTINE. Yes. I couldn't agree with you more.

I would submit that the operating experience of the plants is giving us some of those warning signs. The Davis-Besse event, Rancho Seco event, San Onofre event—to some extent, all of these events are showing that our assumptions about how equipment would perform and the availability of equipment, in some respects are misplaced.

The number of equipment malfunctions and failures, the number of loss of safety systems in those events, is far more than we would have expected.

Now, we did have some forward-looking programs to try and spot those problems early. I think that is the answer. I think that is much of what you suggested.

We have had some team inspection programs that have examined plants to look at particular safety systems and we saw the same kind of vulnerability we saw resulting in serious operating events at Davis-Besse, for example, at these other plants.

We did one of those at Turkey Point, and sure enough, we found some of those same kinds of vulnerabilities.

By identifying them early and by getting the utility to focus attention on them, we may prevent the kind of serious operating events or worse, an accident, that might otherwise occur.

That is why we need to do more of those forward-looking programs.

Unfortunately, with the budget reductions in Fiscal Year 1987, the forward-looking programs are the ones that are getting cut back.

We ask OMB for somewhere in the neighborhood of $460 million for Fiscal Year 1987 to do some of these forward-looking things.

OMB gave us $405 million. I think we need more than $405 million. I think the Fiscal Year 1987 level is cutting into the muscle, as Fred said.

I think somewhere in the neighborhood of $430 million with a specific directive that the additional money be used for these kinds of forward-looking efforts would do a lot to help improve our ability to assure safety and, as you say, just make sure the likelihood of one of these accidents is very, very low and kept very low.

Mr. ZECH. I agree with you, too. We should always be vigilant for any problems that might occur in our plants.

I visited 59 plants in the United States now at our commercial nuclear power operations.

It is my judgment that today our commercial nuclear power reactors are being operated safely. There is a great deal of attention to detail.

There is a great effort to focus on safety. It is paramount. It has certainly been my impression as I have visited our plants.

It does not mean that there is not room for improvement, but you should be assured that the reactor operators in our commercial nuclear power industry are acting very responsibly and are keeping safety paramount.

Just to answer briefly your question about whether we have enough funds for inspection and enforcement and so forth, it is true that we only monitor and audit a very small percentage of our construction and operational testing activities.

That is the way the NRC and the AEC have been set up for many years.

We monitor, I would guess, one percent, two percent, three percent, not much more than that. That is what we do.

Now, if that is to be changed, it would be a significant change in the way we do business.

It is my view that the funds have been provided, that what we have requested under our current systems of operations is sufficient.

We should be able to do the job with what we have. We should not be callous about this Chernobyl incident. It was a very significant incident. It should have a sobering effect on all of us involved with nuclear power.

We should look very carefully to see what changes, if any, need to be done. I think it is our responsibility to do that.

I assure you that I will give that my full attention in these next weeks and months, and that is exactly what we on the NRC intend to do.

If any changes need to be made, if any additional funds that have been requested as far as I am concerned, we will make that request to the Congress and looking out for the number one mission we have and that is the protection of the public health and safety.

Mrs. **Smith**. Now, thank you all.

You commented that the plants around the Great Lakes are operating very well, that perhaps 20 percent of our plants are operating less satisfactorily than we might wish.

I would like to ask you specifically about the two we have in Nebraska, OPPD plant at Fort Calhoun and also OPPD plant at Brownville.

Mr. **Palladino**. I am sorry. I didn't hear the question.

Mr. **Zech**. I am sorry, Mrs. Smith. Those are two I have not visited yet, but I promise you I will do that very soon.

Mrs. **Smith**. You mean you can't give me an answer this morning?

Mr. **Zech**. I am saying personally I have not been able to visit those plants. I am sure we can give you an answer from our staff.

Mrs. **Smith**. Mr. Denton, do you have an answer?

Mr. **Denton**. I have given you my own opinion, which may not be the full agency opinion, and the inspectors might have other views, but I think that is a good comparison of two municipal owned power companies.

They both, as I understand, are owned by government agencies, in effect.

Mrs. **Smith**. Yes.

Mr. **Denton**. I think Fort Calhoun is definitely operating above average. Cooper is not.

We have talked about that the last time we met with the regional administrator; could we pinpoint what is it about the situation there where they are both in the same state, they are both, in effect, owned by local agencies and one seems to be achieving much better performance than the other.

I can't go much beyond that answer other than to say it is the general perception our staff has.

I think we give Fort Calhoun a star. We wouldn't give Cooper the star.

Mr. **Palladino**. May I suggest we add to that for the record from our Systematic Assessment Program?

That might be helpful.

[The information follows:]

APR 10 1985

In Reply Refer To:
Docket: 50-298/85-06

Nebraska Public Power District
ATTN: J. M. Pilant, Manager, Technical
Staff-Nuclear Power Group
P. O. Box 499
Columbus, Nebraska 68601

Gentlemen:

This forwards the report of the Systematic Assessment of Licensee Performance (SALP) Board for the Cooper Nuclear Station. The SALP Board met on March 12, 1985, to evaluate the performance of the Cooper Nuclear Station for the period July 1, 1983, through January 31, 1985. The performance analyses and resulting evaluations are documented in the enclosed SALP Board Report.

The performance of your facility was evaluated in the selected functional areas identified in Section IV of the enclosed SALP Board Report.

The overall performance of the Cooper Nuclear Station has been satisfactory. There are questions, however, raised by this report concerning the commitment and effectiveness of management at the district level. The fact that performance did not improve significantly in any functional area suggests this.

The Performance Appraisal Team (PAT) inspection occurred late in the SALP evaluation period. As a result, regional followup to PAT potential enforcement findings was not complete at the end of the SALP period. Violations written as a result of regional followup to the PAT inspection will be considered in the next SALP report.

Unless you specifically request it, a management meeting will not be scheduled for this reporting period. Any request for a management meeting should be made in writing within 10 days of receipt of this report.

Any comments which you may have regarding our evaluation of the performance of your facility should be submitted to this office within 30 days of the date of issue of this SALP Board Report. Your comments, if submitted, and our disposition of them, will be issued as appendices to the SALP Board Report.

Comments which you may submit are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act, PL 96-511.

Should you have any questions concerning this letter, we shall be pleased to discuss them with you.

Sincerely,

[illegible] signed by
[illegible]

Robert D. Martin
Regional Administrator

Enclosure:
Appendix - SALP Board Report 50-298/85-06
  Attachment 1 - NRR Licensing Input
  Attachment 2 - AEOD Input

Paul V. Thomason, Division Manager
  of Nuclear Operations
Cooper Nuclear Station
P. O. Box 98
Brownville, Nebraska 68321

bcc to DMB (IE40)

bcc distrib. by RIV:
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*Section Chief (RPB1/A)
*EP&RPB
*RIV File
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R. P. Denise, DRSP
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Chmn. Palladino (MS: H-1144)
Comm. T. M. Roberts (MS: H-1149)
Comm. L. W. Zeck (MS: H-1149)
*M. Murphy, SPES

*w/766

SALP BOARD REPORT

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE

50-298/85-06
Nebraska Public Power District

Cooper Nuclear Station (CNS)
July 1, 1983 through January 31, 1985

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PDR ADOCK 05000298
G PDR

I. INTRODUCTION

The Systematic Assessment of Licensee Performance (SALP) program is an integrated NRC staff effort to collect available observations and data on a periodic basis and to evaluate licensee performance based upon this information. SALP is supplemental to normal regulatory processes used to ensure compliance to NRC rules and regulations. SALP is intended to be sufficiently diagnostic to provide a rational basis for allocating NRC resources and to provide meaningful guidance to the licensee's management to promote quality and safety of plant construction and operation.

An NRC SALP Board, composed of the staff members listed below, met on March 12, 1985, to review the collection of performance observations and data to assess licensee performance in accordance with the guidance in NRC Manual Chapter 0516, "Systematic Assessment of Licensee Performance." A summary of the guidance and evaluation criteria is provided in Section II of this report.

This report is the SALP Board's assessment of the licensee's safety performance at Cooper Nuclear Station for the period July 1, 1983 through January 31, 1985.

The SALP Board was composed of the following members of the NRC staff:

E. H. Johnson, RIV, Chairman
R. L. Bangart, RIV
L. J. Callan, IE
D. DuBois, RIV
K. Kniel, NRR
E. Sylvester, NRR

Other attendees who participated in all or part of the Board's deliberations were:

R. J. Everett, RIV
M. E. Murphy, RIV
B. Murray, RIV
W. C. Seidle, RIV
C. Hackney, RIV

II. CRITERIA

Licensee performance is assessed in selected functional areas, depending upon whether the facility is in a construction, preoperational, or operating phase. Each functional area normally represents areas significant to nuclear safety and the environment, and are normal programmatic areas. Some functional areas may not be assessed because of little or no licensee activity or lack of meaningful observations. Special areas may be added to highlight significant observations.

One or more of the following evaluation criteria were used to assess each functional area.

1. Management involvement and control in assuring quality.
2. Approach to resolution of technical issues from a safety standpoint.
3. Responsiveness to NRC initiatives.
4. Enforcement history.
5. Reporting and analysis of reportable events.
6. Staffing (including management).
7. Training effectiveness and qualification.

However, the SALP Board is not limited to these criteria, and others may have been used where appropriate.

Based upon the SALP Board assessment, each functional area evaluated is classified into one of three performance categories. The definitions of these categories are:

Category 1. Reduced NRC attention may be appropriate. Licensee management attention and involvement are aggressive and oriented toward nuclear safety; licensee resources are ample and effectively used so that a high level of performance with respect to operational safety or construction is being achieved.

Category 2. NRC attention should be maintained at normal levels. Licensee management attention and involvement are evident and are concerned with nuclear safety; licensee resources are adequate and are reasonably effective so that satisfactory performance with respect to operational safety or construction is being achieved.

Category 3. Both NRC and licensee attention should be increased. Licensee management attention or involvement is acceptable and considers nuclear safety, but weaknesses are evident; licensee resources appear to be strained or not effectively used so that minimally satisfactory performance with respect to operational safety or construction is being achieved.

III. SUMMARY OF RESULTS

1. Strengths

NPPD management and staff have been responsive to NRC requirements and licensing concerns. Major station design changes in progress or previously completed have indicated management's attention to upgrading CNS to be a more efficient, reliable, and safer plant.

Strong ALARA controls were evident during the performance of all work activities. The CNS quality assurance group played a major role in the identification of quality problems and ensuring adequate resolution of discrepancies. The site engineering department has increased the depth of their evaluations, particularly in the areas of problem analysis, resolution of generic problems, and specificity of written reports and records.

The NPPD corporate office and site organizations were restructured during this SALP period. As a result of the reorganization, improvements occurred in staffing size, quality of leadership, personnel motivation and morale, and quantity of the work effort, although many improvements remain to be realized.

2. Weaknesses

Manpower losses of experienced members of the professional and technical staffs continue at CNS. Office and working areas are near capacity and may be insufficient to handle increasing staffing needs. Many formal programs have not been developed in areas such as training, quality control, and trend analysis. The licensee does not have their own plant-specific simulator. The NRC has identified problems in the control of surveillance testing and has questioned the technical adequacy of the surveillance test program at CNS.

3. Performance Category

A summary of the licensee's performance, as determined during the SALP Board meeting, is shown in the table below. Also included is the performance category from the previous evaluation period.

	Previous Assessment Period	This Assessment Period
Functional Area	July 1, 1982 - June 30, 1983	July 1, 1983 - January 31, 1985
A. Plant Operations	2	2
B. Radiological Controls*		2
1. Radiation Protection	2	
2. Chemistry/Radiochemistry and Confirmatory Measurements	2	
3. Radwaste Management, Effluent Releases, and Effluent Monitoring	not inspected	

Functional Area	Previous Assessment Period July 1, 1982 - June 30, 1983	This Assessment Period July 1, 1983 - January 31, 1985
4. Transportation Activities and Solid Radwaste	1	
5. Environmental Monitoring	1	
C. Maintenance	1	2
D. Surveillance	1	2
E. Fire Protection	1	1
F. Emergency Preparedness	2	2
G. Security Safeguards	2	2
H. Refueling/Major Outages	1	1
I. Training	3	3
J. Licensing Activities	1	1
K. Quality Programs and Administrative Controls Affecting Quality**	2	2

*This category was shown divided into five subcategories since onsite/offsite responsiblities are divided among separate groups.

**This category was divided into two individual categories in the previous SALP report and each was given a Performance Category 2.

The total NRC inspection effort during this SALP evaluation period consisted of 48 inspections involving a total of 3669 hours onsite by NRC inspectors. NRC inspection reports issued during this assessment period were:

- 83-16 through 83-34
- 84-01 through 84-27
- 85-01 through 85-04

IV. PERFORMANCE ANALYSIS

A. Plant Operations

1. Analysis

This area was inspected by the NRC Senior Resident Inspector (SRI) on a continuing basis and by the NRC Performance Appraisal Team (PAT). The inspections included observation and review of plant operations, shift manning, equipment status controls, shift records, crew training, reportable occurrences; independent verification of system lineups and condition of operability; and verification of licensee adherence to administrative controls, plant procedures, and the requirements established in the CNS Operating License and Technical Specifications.

The three violations listed below were attributed to activities in the functional area of plant operations:

- Failure to correctly perform Surveillance Procedure 6.2.4.1, Attachment A, page 6 of 15, check 1, "Reactor Recirculation System Jet Pump Operability." (Severity Level IV, 8404)
- Failure to correctly perform Surveillance Procedure 6.2.4.1, Attachment A, page 13 of 15, "Standby Liquid Control System." (Severity Level IV, 8415)
- Failure to follow procedure during the performance of Surveillance Procedure 6.3.9.4, "Main Steam Isolation Valves Operability Test." (Severity Level IV, 8418)

The three Licensee Event Reports (LER) listed below were attributed to activities in this functional area:

- A reactor trip was caused by automatic closure of all main steam isolation valves (MSIV). The MSIVs closed when steam line pressure dropped to less than 825 psig while the reactor control mode switch was in the RUN position. The low pressure condition was caused by a loss of both reactor recirculation system pumps while operating at 79 percent power. (LER 84-02)
- Unmonitored release of radioactive liquid effluent due to a station operator improperly performing the monitoring system valve lineup. (LER 84-08)
- A reactor trip was caused by the existence of a high temperature condition in the steam tunnel. (LER 84-10)

During this assessment period, the licensee exhibited strong management oversite of plant operations, and placed emphasis upon improving the performance level in this functional area. The licensed operator training and retraining programs improved significantly in quality and quantity. Various plant modifications in progress include installation of the Safety Parameter Display System, a new plant computer, development of new symptomatic emergency procedures, and a control room design review. Once completed these will enhance operator effectiveness and plant safety. A control room supervisor (CRS) position was established for each operating crew. The CRS will provide additional oversite of plant operations, adherence to Technical Specification requirements, and review of plant logs and surveillance test data. The CRS reports directly to the shift supervisor. The shift supervisor and CRS each hold an NRC Senior Reactor Operator (SRO) license.

There is a need for improved attention to detail when recording, interpreting, and reviewing Technical Specification surveillance data. Also, violations involving failure to follow procedures needs greater managerial attention. Decision making by shift supervisory personnel was not always strong in that the concept of believing the worst indication was not practiced without consultation with higher levels of plant management. Administrative controls of jumpers and bypasses and the use of "red arrows" was marginal.

Improvements to items identified during the previous SALP period include an increase in the development of specific maintenance related procedures, increased visibility and usage of station procedures, increased recruitment and retention of operations personnel, improvement of reactor operator training, and development of the symptomatic emergency procedures. Significant improvement is still needed in the areas of procedure adherence, familiarity with the CNS Technical Specifications, administrative controls of "red arrows", modernization of station procedures, and meeting the commitment to staff six operating crews.

The licensee's reply to the previous SALP report was reviewed, and it was determined that licensee actions were responsive to the NRC recommended licensee actions. However, it was apparent that licensee corrective actions were not timely enough and lacked sufficient strength to prevent recurrence of similar problems identified during this assessment period.

2. Conclusions

The licensee is considered to be in Performance Category 2 in this area.

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should remain at its current level.

b. Recommended Licensee Actions

Licensee management attention should be increased in the areas of Technical Specification familiarity, procedure adherence, improving the quality of station procedures, updating administrative controls which directly affect operations personnel, and crew supervisory responsibilities and decision making. Licensee management should continue monitoring progress in the human factors studies and implementation of plant hardware and procedure changes resulting from those studies. Also, licensee management is encouraged to strengthen their commitment to design and construct a plant specific simulator.

B. Radiological Controls

1. Analysis

Ten inspections were conducted during the assessment period regarding radiological controls by region-based radiation specialist inspectors. These ten inspections covered the following areas: radiation protection - normal operations; radiation protection - recirculation pipe replacement outage; radwaste management, effluent releases, and effluent monitoring; chemistry/radiochemistry and confirmatory measurements; and radioactive material transportation activities. Eight violations were identified:

- Failure to Leak Test Sealed Radioactive Sources. (Severity Level IV, 8325)
- Failure to Evaluate Personnel Dosimetry Results. (Severity Level IV, 8325)
- Failure to Maintain Survey Records. (Severity Level IV, 8325)
- Failure of Radiation Worker to Wear Prescribed Personnel Monitoring Dosimetry. (Severity Level IV, 8325)
- Failure to Perform Surveys to Properly Evaluate Radiation Hazards. (Severity Level V, 8325)

- Failure to Properly Control Work Activities in High Radiation Areas. (Severity Level IV, 8406)
- Failure to Follow Radiation Control Procedures. (Severity Level IV, 8406)
- Failure of Radwaste Shipment to Comply with Radiation Level Limits. (Severity Level III, 8412)
  Note: This was identified by the State of Nevada

The one LER listed below was attributed to activities in this functional area:

- Unmonitored release of gaseous effluent at the elevated release point (ERP), due to a chemistry technician not following procedure when changing out in line filters. (LER 84-06)

a. Radiation Protection

This area was inspected once during normal plant operations and three times regarding the recirculation pipe replacement outage.

The licensee's person-rem for calendar year 1983 was 1,293, as compared to the BWR national average of 1,093. Approximately 700 person-rems for 1983 was the result of weld overlay work to repair cracks in the recirculation loops and other piping.

During calendar year 1984, 743 person-rem was expended. The 1984 person-rems included 588 as a result of recirculation pipe replacement and 155 due to normal plant operations. The licensee's person-rem average between 1975 and 1983 was 467 which is well below the national BWR average of 834 for the same time period. Much of the licensee's success in maintaining their person-rem below the national average is attributed to: (1) a good reactor coolant water chemistry program, and (2) a low incident of fuel leakage. These two factors have resulted in lower radiation levels throughout the plant when compared to other BWRs.

A new chemistry and health physics supervisor (radiation protection manager) was designated during the assessment period. Even though the new chemistry and health physics supervisor met the minimum literal interpretation of Regulatory Guide 1.8, the NRC expressed concerns that management had not demonstrated a commitment for program excellence by the selection of a professional health physicist with health physics experience at commercial power reactors. In response to the NRC concerns, the licensee committed to having the work activities of the present chemistry and health physics supervisor reviewed and monitored by a qualified person that had previously held the position as radiation protection manager until such time as the new individual attains the necessary power reactor experience.

The licensee's radiation protection staff includes 17 individuals which is considered to be adequate to support normal plant operations. A well defined training and qualification program has not been established for the radiation protection staff. The licensee experienced a turnover rate of less than 5 percent in the radiation protection staff in 1984. About 40 percent of the radiation protection technicians are considered qualified in accordance with American National Standards Institute (ANSI) Standard N18.1-1971.

The licensee implemented a well defined radiation protection program associated with the recirculation pipe replacement outage. The licensee supplemented their permanent plant radiation protection staff with an adequate number of ANSI qualified contractor radiation protection technicians. A detailed radiation protection plan was developed for this outage. Within the radiation protection organization, a positive attitude exists regarding radiation protection. The licensee has implemented an aggressive program to assure radiation controls are implemented. A good housekeeping program was evident.

The licensee had implemented several good ALARA practices associated with the recirculation pipe replacement work. These included: (1) chemical decontamination of piping inside surfaces prior to beginning pipe removal procedures, (2) an extensive mockup training program, (3) use of remote video systems in the drywell to observe work activities, and (4) a new pipe preconditioning program in order to reduce the buildup of activated corrosion products after reactor startup. A comprehensive ALARA program had been established for the pipe replacement outage. However, the licensee does not have a comprehensive ALARA program for normal plant operations.

The licensee has shown a willingness to respond to NRC identified initiatives. However, their program indicates a weakness in that the violations identified by the NRC should have been identified within their organization. The licensee's approach to major problems concerning radiation protection is generally sound, but inattention to detail has resulted in several minor violations.

b. Chemistry/Radiochemistry and Confirmatory Measurements

Two inspections of chemistry/radiochemistry and confirmatory measurement activities were performed during the assessment period. These inspections involved onsite confirmatory measurements with the RIV mobile laboratory. Both inspections resulted in good agreement between the NRC's and the licensee's measurements. The chemistry/radiochemistry staff has experienced a low turnover rate during the past several years. As a result, a stable program exists with an adequate number of experienced technicians. A well organized training program has not been developed for the chemistry/radiochemistry area.

The second inspection of this area also included, for the first time, confirmatory measurements on the licensee's whole body counting system. Problems identified included: (1) low analytical results for activity greater than two body burdens of Co-60, (2) high results for I-131 in the thyroid, and (3) Co-60 in the lungs has a masking effect over I-131 in the thyroid. Well established regulatory criteria does not currently exist for whole body counting systems, but the licensee agreed to review the problems identified with their program.

c. Radwaste Management, Effluent Releases, and Effluent Monitoring

This area was inspected twice during the assessment period. No particular problems were noted. This area was assigned a Performance Category 1 in the 1981-82 SALP report. As a result, this area was not inspected during the 1982-83 assessment period. The licensee's past performance has been consistently high in this area. The effluent sampling and analysis activities are well defined with only minor changes occurring in the program. There has been little turnover in the past several years regarding personnel responsible for performing effluent analyses. However, the same level of stability was not observed at the utility operator level for operations personnel responsible for operating the various radwaste systems. A well defined training program does not exist for the radwaste utility operators.

d. Transportation Activities and Solid Radwaste

This area was inspected twice during the assessment period. One violation regarding radiation levels that exceeded regulatory limits was identified at the low-level waste burial site by state inspectors. No other problems were identified in this area.

e. Environmental Monitoring

This area was not inspected during the assessment period. The licensee was considered to be in Performance Category 1 in this area for the 1982-83 SALP report. Accordingly, the NRC inspection effort was reduced during this assessment period. The licensee's performance in this area has been consistently high. A well managed program has been in operation for several years. The next environmental monitoring inspection is scheduled for early 1985.

2. Conclusions

The licensee continues to maintain one of the lowest person-rem exposure histories among BWRs. This success is mainly attributed to a good reactor coolant chemistry program to prevent the buildup of activated corrosion products and a low rate of fuel leakage. The licensee established a new pipe preconditioning program which should also contribute to maintaining low in-plant radiation levels after reactor startup. An adequate radiation protection program was established for the recirculation pipe replacement outage. This included the development of an effective radiation protection plan, the addition of temporary contract health physics technicians, a detailed ALARA program, and close monitoring of work activities.

The size of the permanent radiation protection staff is considered adequate to support plant operations. However, a strong management commitment to ensure quality in this staff was not evident when the licensee selected a person who was not a professional health physicist with commercial reactor health physics experience to fill the vacant radiation protection manager position, as recommended by ANSI Standard N18.1-1971. The licensee's program is considered adequate regarding major program activities; however, inattention to program details has resulted in several violations.

No significant problems were identified in the functional areas of chemistry/radiochemistry, and confirmatory measurements; radwaste management, effluent releases, and effluent monitoring; solid radwaste; and environmental monitoring. The licensee's programs for these areas were adequate regarding management oversight, resolution of technical issues, enforcement history, and staffing.

Well defined training programs had not been established for any of the functional areas included in radiological controls.

The licensee is considered to be in Performance Category 2 in this area.

3. Board Recommendations

a. Recommended NRC Actions

The level of NRC inspection effort in this area should be consistent with the routine inspection program.

b. Recommended Licensee Actions

Increased management attention is needed to assure that an adequate level of technical expertise is maintained in the radiation protection staff. Management should assure that the ALARA program implemented for the pipe replacement outage is continued during normal plant operations. Management attention is needed to assure that adequate training programs are developed and implemented for all segments of the functional area of radiation protection.

C. Maintenance

1. Analysis

This area was inspected by the SRI on a continuing basis, periodically by region-based NRC inspectors, and by the PAT team. The inspections included observations of work activities performed by the maintenance organization and reviews of administrative controls, schedules, procedures, and records.

One violation listed below was attributed to activities in the functional area of maintenance:

- Failure to have procedures for (1) use and calibration of crimping tools for safety-related cable termination lugs and splices; (2) use in determining lubrication requirements, bearing life, and component part life less than 40 years; and (3) use to assure lubrication or part replacement to prevent voiding of equipment qualification and degradation of safety-related equipment during plant life. (Severity Level IV, 8411)

The five LERs listed below were attributed to activities in this functional area:

- Pressurization of the suppression chamber air space occurred due to a stuck open high pressure coolant injection (HPCI) line vacuum breaker. (LER 83-14)
- A reactor trip resulted from an inadvertant loss of a reactor feedwater pump during instrument and control (I&C) department troubleshooting operations which were not properly controlled. (LER 84-03)

- Leakage of primary coolant from a crack in the weld of an elbow-to-socket joint located on the "B" Reactor Recirculation Pump discharge drain line isolation valve. The probable failure mechanism was an inadequate welding operation. (LER 84-05)

- Both trains of the Standby Gas Treatment System were made inoperable when their associated charcoal filtration beds were deluged by water from the fire protection system. The initiating event was the shearing of a fire hydrant by a construction bulldozer. (LER 84-07)

- Failure of a diaphragm in the HPCI turbine overspeed trip auto reset control valve actuator. The cause was attributed to end of component life. (LER 84-11)

During this assessment period, management has continued to aggressively identify, repair, and restore plant equipment that has required repair or routine preventive upkeep. However, there were problems noted in: procedures for performing safety-related work activities and calibrating measuring and test equipment; oversight of activities performed by contractors; documentation associated with some work performed by nonlicensee personnel; failure to have approved training programs for maintenance personnel, and controls of vendor documents and shop guides that are used in safety-related maintenance activities.

Four technicians, two lead technicians, and one engineer have resigned from the I&C department during this period. This high attrition rate has left the licensee with only four experienced I&C technicians.

The previous SALP report identified weaknesses in the areas of procedure adherence and lack of procedures for performing activities requiring meticulous controls or more than reliance on memory. Improvements have occurred in the development of numerous maintenance related procedures that did not previously exist.

The SRI reviewed the licensee's reply to the previous SALP report and determined that their actions were responsive to the NRC recommended licensee actions. However, the development and implementation of formalized training for maintenance personnel has not been completed.

2. Conclusions

Due to the absence of an aggressive program to control all safety-related vendor supplied information and manuals, to control shop guides in the same manner as plant procedures, and to establish formalized classroom training for maintenance personnel, the licensee's overall performance has decreased in this functional area.

The licensee is considered to be in Performance Category 2 in this area.

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should remain at a normal level.

b. Recommended Licensee Actions

Licensee management attention should be increased in the areas of maintenance procedures development and controls of vendor documents and shop aids used directly or indirectly in the performance of safety-related maintenance or preventive maintenance activities. The licensee should establish and maintain a formal training program for maintenance and I&C personnel. Immediate attention must be given to the the high attrition rate of I&C personnel.

D. Surveillance

1. Analysis

This area was inspected by the SRI on a continuing basis, periodically by region-based NRC inspectors, and by the PAT team. The inspections included observations of surveillance activities and reviews of applicable administrative controls, schedules, procedures, and records.

The two violations listed below were attributed to activities in the functional area of surveillance:

- The licensee failed to adequately demonstrate operability of station batteries. (*)

- The licensee failed to have procedures for conducting battery charges. (*)
  * The above violations were evaluated as a Severity Level III violation. The NRC has proposed an imposition of civil penalty for these violations. (EA 84-132)

The four LERs listed below were attributed to activities in this functional area:

- During the performance of local leak rate testing (LLRT) 20 penetrations were found to be leaking excessively. (LER 83-13)
- Reactor Recirculation Pump "A" high pressure trip relay, RR-REC-K48A, failed to energize during the performance of Surveillance Procedure 6.2.8.3. The failure cause could not be determined. (LER 83-15)
- Reactor Vessel Water Level high level trip switch, NBI-LIS-101B, failed to reset during the performance of Surveillance Procedure 6.1.9. The apparent cause of switch failure was end-of-life. (LER 84-01)
- Failure of the Reactor Core Isolation Coolant (RCIC) Turbine governor control system due to a broken solder connection. (LER 84-04)

During this assessment period, the licensee has maintained an aggressive attitude towards timely completion of the plant surveillance program. The surveillance coordinator, a highly experienced former CNS shift supervisor, ensured that the surveillance program was well planned and implemented. Also, the licensee satisfactorily completed major periodic surveillances such as the Primary Containment Integrated Leakage Test (ILRT), Reactor Vessel In-Service Leak Test, and Reactor Recirculation System Inservice Inspection (ISI) Program. The ISI program was performed primarily as a result of several IE Bulletins which discussed Intergranular Stress Corrosion Cracking (IGSCC) in large diameter stainless steel recirculation system piping. The licensee was replacing recirculation system piping and other reactor coolant pressure boundary piping during the refueling outage in progress at the end of this evaluation period.

Problems identified in this area included inadequate repair or operational degradation of primary containment isolation valves and procedures which lacked specificity with regards to delegation of surveillance monitoring responsibilities. The violations associated with the station batteries indicated a poor understanding of battery technology in preparing procedures and establishing record requirements for the batteries, and resulted in an escalated enforcement action. The above conditions brought into question the technical adequacy of the overall surveillance test program.

The previous SALP report indicated problems with excessive primary containment leakage. Excessive primary containment leakage is still discovered during the performance of LLRT; however, the licensee has established a trend analysis history of problem valves and have used it to initiate rework, replacement, and/or modification of the problem valves. The licensee has contracted repair/rework services for some primary containment isolation valves from the respective vendors and subsequent LLRTs of those specific valves indicated a marked improvement.

The licensee's reply to the previous SALP report appeared to respond to the NRC recommended actions. Improvement has occurred in the areas of evaluation and correction of containment isolation valve leakage, establishment of a full-time outage coordinator, and training of selected maintenance personnel on Limitorque valves.

2. Conclusions

The licensee's overall performance has decreased in this functional area due to the identification of technically inadequate surveillance procedures, lack of procedures for performing safety-related surveillance activities, and continued excessive primary containment leakage problems.

The licensee is considered to be in Performance Category 2 in this area.

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should be increased with emphasis in the area of surveillance program adequacy and implementation.

b. Recommended Licensee Actions

Licensee management attention should be increased in the areas of surveillance procedure content and technical accuracy. The licensee should continue to closely monitor and trend LLRT results. Also, management should consider purchasing a valve lapping, cutting, and grinding machine that is capable of providing better repairs of containment isolation valves.

E. Fire Protection

1. Analysis

This area was inspected on a continuing basis by the SRI and periodically by region-based NRC inspectors. The inspections included observation and review of the fire protection/prevention program, housekeeping/cleanliness controls, storage and protection of quality station documents, and training of the fire brigades.

The four violations listed below were attributed to activities in the functional area of fire protection:

- The licensee failed to include Fire Door P105 in Surveillance Procedure 6.4.5.1, "Fire Protection System Monthly Inspection." (Severity Level IV, 8320)
- Surveillance and calibration records, maintained in temporary storage for as long as 2 years, were not afforded fire protection. (Severity Level V, 8324)
- Failure to use flame resistant materials for covering materials stored out-of-doors. (Severity Level V, 8422)
- Fire doors do not operate properly and are not maintained adequately. (Severity Level V, 8423)

No LERs were identified in this functional area.

During this assessment period, the licensee expended considerable effort to upgrade the fire protection system to meet station needs and 10 CFR Part 50, Appendix R, requirements. The new fire protection clean water storage tanks, fire pumps, and additional distribution piping and hydrants were placed in service. Housekeeping conditions at the plant were excellent and demonstrated a high level of management concern. Administration of the fire protection program, controls, and surveillance testing are strongly supported by management. The previously listed violations indicate a need for improvement in the areas of records storage, protection of parts and materials stored outdoors, and greater attention to the care and maintenance of fire doors.

A review of the previous SALP report and the licensee's response to that report indicated that management has continued to maintain an aggressive attitude in this area and has provided support as required.

2. Conclusions

The licensee is considered to be in Performance Category 1 in this area.

3. Board Recommendations

a. Recommended NRC Actions

The level of NRC inspection effort should remain about the same with emphasis placed on the licensee's completion of 10 CFR Part 50, Appendix R, commitments.

b. Recommended Licensee Actions

Licensee Management is encouraged to continue its involvement in upgrading the fire protection system to meet 10 CFR Part 50, Appendix R, requirements. Also, licensee management should ensure that fire protection of records and other stored materials is adequate and that fire doors are properly maintained. The licensee should consider instituting hands-on training for fire brigade members which includes a smoke filled, dark, poorly ventilated, and maze type environment which would most probably be encountered during a significant plant fire.

F. Emergency Preparedness

1. Analysis

During the assessment period, five emergency preparedness inspections were conducted. The first inspection, conducted August 1-5, 1983, was a review of the emergency preparedness program in the areas of emergency detection and classification, protective action decisionmaking, training, emergency notification and communications. No violations were identified, but concerns were identified with regard to shift supervisor independence in making offsite recommendations and incorporation of requirements for notification of offsite authorities in the NPPD emergency plan and procedures. These concerns were resolved during the assessment period.

The next inspection was conducted during the period August 21-26, 1983. This inspection reviewed the licensee's programs for informing the public of initial actions to be taken in an emergency and emergency worker radiation protection. The licensee's programs in these areas were found to be adequate; however, four concerns were identified. These concerns included failure to distribute Public Information brochures to a nearby state park, maintenance of the brochure mailing list, lack of a system to contact corporate personnel assigned emergency functions, and omission of titles and telephone numbers in the corporate emergency plan. The resolution of some of these concerns was slow during the period.

During the period October 3-7, 1983, an inspection was conducted which reviewed emergency preparedness training, emergency preparedness program review and meteorological instrumentation surveillance. As a result of this inspection, a Notice of Violation was issued for failure to conduct a 12-month program review that was in full compliance with the requirements of 10 CFR Part 50.54(t). This violation is discussed in Section K. In addition, it was determined that the interface between the emergency preparedness group and the training department was not adequately defined. Both of these problems were satisfactorily addressed by the licensee prior to the end of the assessment period.

On March 15, 1984, an inspection was conducted to review the annual emergency preparedness exercise objectives in conjunction with a review of the offsite agencies' exercise objective by the Federal Emergency Management Agency, Region VII. The licensee's objectives were determined to be adequate in regard to exercising the emergency plan in accordance with NRC regulatory requirements.

Implementation of the licensee's emergency plan and procedures was reviewed during an inspection conducted May 14-17, 1984. The inspection included observation of a full scale emergency exercise at CNS on May 16, 1984. Several areas were identified in which improvements could be made; however, good performance was observed during the exercise in effective use of control room personnel, emergency coordinator leadership and security for the emergency operations facility. During the exercise no violations were identified. During the assessment period several organizational changes were made at CNS in the areas of operations, training and emergency preparedness which appeared to strengthen the emergency preparedness program. The CNS emergency preparedness program was considered to be adequate during the last part of the assessment period, in contrast to the first part of the period in which staffing was considered to be a problem due to the lack of a full time emergency planning coordinator onsite.

2. Conclusion

The licensee has maintained an adequate emergency preparedness program during the period and demonstrated the capability to protect the health and safety of the public in the event of an accident at CNS. In general the licensee's management controls and resolution of NRC concerns have been effective and timely during the period. However, the licensee's response to some of the NRC concerns identified above required additional NRC effort toward obtaining acceptable resolutions.

The licensee is considered to be in Performance Category 2 in this area.

3. Board Recommendations

a. Recommended NRC Action

The NRC inspection effort in this functional area should continue at a normal level.

b. Recommended Licensee Action

The current level of management attention to implementation of the emergency response and preparedness program should continue. Additional management attention should be given to the prompt resolution of NRC identified concerns.

G. Security Safeguards

1. Analysis

The physical security staff performed five inspections during this SALP period. Only two of these inspections were related to security. Two violations and four deviations were identified during these inspections.

The violations and deviations attributed to this functional area are:

- A violation concerning access control for vehicles (8413-08)
- A deviation concerning preventive maintenance (8413-03)
- A deviation concerning emergency security power (8413-04)
- A deviation concerning security post orders (8413-05)
- A deviation concerning contractor background checks (8413-07)
- A violation concerning guard examinations (8428-01)

The initial phase of this reporting period was one of transition for the licensee. Past problems in the areas of management control, an aging security plan and lack of support staff, indicated a need for strong management attention. A new security management team now in charge at the site seemed to recognize the situation and have begun to take some positive steps to resolve security program deficiencies. Currently, there is an effective channel of communication between the site management responsible for security and the regionally based NRC security inspectors.

2. Conclusions

The licensee has taken several positive actions to resolve the underlying weaknesses of the security program. The site security plan has been reviewed and several key revisions have been made. Site security management reflects a dedication to improve the security program.

The licensee is considered to be in Performance Category 2 in the security area. The positive steps initiated by the licensee should enhance the security program but will take time to fully implement.

3. Board Recommendations

a. Recommended NRC Action

The level of NRC inspection effort during this revitalization period should be increased to exceed the minimal program prescribed in the security and safeguards operational procedures.

b. Recommended Licensee Action

Continue to pursue aggressively the corrective steps already taken and the planned revisions to the security organization and support systems.

H. Refueling/Major Outages

1. Analysis

This area was inspected by the SRI and region-based NRC inspectors. Inspections included outage planning, preparation for refueling, refueling activities, plant startup following refueling, and physics testing following refueling.

One violation listed below was attributed to activities in the functional area of refueling:

- Loading fuel into the reactor core with control rods not fully inserted. (Severity Level IV, 8331)

No LERs were identified in this functional area.

During this assessment period, the licensee's programs for outage planning, maintenance, plant modifications, refueling preparations, conduct of refueling, post-refueling surveillance testing, and restoration of systems prior to plant startup were found to be well planned, performed, and documented. The violation listed above was caused by a combination of administrative errors and procedural deficiencies. The violation was discussed in the previous SALP report since it actually occurred during that period. The violation itself was subsequently issued during this assessment period.

The commitments made by the licensee in their reply to the previous SALP report were responsive to the NRC recommended actions. The licensee's commitments were completed during this appraisal period.

There were two major outages during this SALP period. The first outage was in progress when this evaluation period started and was completed September 1, 1983. Outage activities included inspection of the main turbine and generator, testing of GE Type 120 VAC HFA relays, weld overlay of reactor recirculation system piping welds because of the identification of IGSCC, performance of a primary containment integrated leak rate test, and refueling operations.

The second major outage commenced September 15, 1984, and was still in progress at the end of this evaluation period. Outage activities included inspection of the main turbine, diesel generators annual inspection, 1B reactor feedwater pump turbine rotor replacement, control room design review, and defueling operations. Also, all reactor recirculation system piping and other attached safety-related piping that form an integral part of the primary system pressure boundary was removed because of IGSCC concerns. Installation of new piping is still in progress. Eight NRC inspections were performed specifically related to the pipe replacement project and included inspections in the areas of radiation protection, QA, procurement, receipt inspection and storage, licensee/contractor project organizations, project planning/scheduling, personnel qualification and training, nondestructive examination, and procedures implementation.

A high level of management attention was dedicated to the pipe replacement project. Emphasis was evident in the areas of program development, procedures adequacy, ALARA controls, quality assurance, and immediate oversight of piping replacement related activities by licensee project management and the project ALARA committee. The pipe replacement project is scheduled to be completed during April 1985.

2. Conclusions

The licensee's overall performance has remained very high in this functional area. The licensee is considered to be in Performance Category 1 in this area.

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should focus less on the activities directly associated with the refueling and more on maintenance and design change activities which also occur during the refueling outage.

b. Recommended Licensee Actions

Licensee management should place more emphasis on maintenance activities and station design changes that occur in conjunction with the refueling outage.

I. Training

1. Analysis

This area was inspected by the SRI, region-based inspectors, and the PAT team. The inspections included observation and review of the licensee's training plans and procedures, licensed operator training, operator requalification training, and training of nonlicensed personnel, including shift technical advisors (STA), contracted personnel, administrative, maintenance, engineering, site, and corporate personnel.

No violations or LERs were attributed to activities in the functional area of training.

During this evaluation period, the licensee has made considerable improvement in the quality and quantity of reactor operator training. The licensee established a formal training department consisting of a training manager, secretary, one educational specialist, an operations training supervisor, and three instructors. Other departmental positions are being filled by a contract with the GE Company. The instructors have been assigned the responsibilities for and conduct of GET, licensed, and nonlicensed training. Also, the training department is developing the overall CNS training program as required to obtain accreditation from the Institute of Nuclear Power Operations (INPO). Available training classroom area was increased through the purchase of two large training trailers.

Improvements have occurred for items identified during the previous SALP period in the areas of site and corporate management support, licensed operator training, requalification training and examinations. maintenance of training records, GET, departmental specialized training, on-the-job training (OJT), decreasing overtime requirements to accommodate training and operational needs, more formalized classroom presentations, and increased instructor and trainee morale and motivation. Weaknesses are still evident in the areas of formalizing training plans, procedures, and lesson plans, nonlicensed departmental training, and size of the licensee's permanent training staff.

Operator licensing examinations were administered several times during the present SALP period. Twelve reactor operator replacement candidates were examined with nine licenses being issued. Fourteen senior operator candidates were examined with seven licenses eventually being issued. Nine reactor operators and senior operators that were licensed were administered requalification examinations by Region IV personnel. Five candidates passed these examinations.

Unannounced site visits were made on two occasions. One, to audit facility grading of facility administered requalification examinations, and the other to proctor the facility administered requalification retake examination for candidates who had previously failed. The facility grading was considered satisfactory.

Comparing the most recent examination results with the results of examinations administered approximately 9 months to 1 year ago shows a noticeable improvement in candidate performance due to the increased awareness of the importance of training.

2. Conclusions

Although the licensee's overall performance has remained the same in this functional area, significant improvements have been initiated and an improving trend has been identified. The licensee is considered to be in Performance Category 3 in this area.

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should remain at an increased level. Particular emphasis should be placed on training program development and implementation, allotment of personnel resources to the training effort, and formalization and implementation of site and corporate nonlicensed training.

b. Recommended Licensee Actions

The licensee is encouraged to expedite filling vacant training staff positions and constructing an onsite simulator. Also, management attention should continue in the areas of training programs development, training of nonlicensed site and corporate personnel, and updating the CNS Training Manual.

J. Licensing Activities

1. Analysis

The NRC Office of Nuclear Reactor Regulation (NRR) has performed an assessment of licensee performance in the area of licensing activities. Refer to Attachment 1 for details of this assessment.

2. Conclusions

As discussed in Attachment 1, the licensee is considered to be in Performance Category 1 in this area.

3. Board Recommendations

a. Recommended NRC Actions

The board has no specific recommendations in this area.

b. Recommended Licensee Actions

The licensee is encouraged to continue the high level of management involvement which is evident in this area.

K. Quality Programs and Administrative Controls Affecting Quality

1. Analysis

This area was inspected by the SRI on a continuing basis, periodically by region-based inspectors, and by the PAT team. The inspections included organization and administration, safety review committees, licensee audits, quality assurance program, facility design changes and modifications, quality control program, procurement activities, and quality materials receipt, handling, and storage.

The nine violations listed below were attributed to activities in the functional area of quality programs and administrative controls affecting quality:

- The minutes of the Safety Review and Audit Board (SRAB) meeting held on March 16, 1982, were not forwarded to the station superintendent within 1 month of the meeting as required by the CNS Technical Specification. (Severity Level V, 8325)
- The control, review, and format of minor design changes (MDC) were not in compliance with applicable administrative procedures. (Severity Level IV, 8326)
- The SRAB failed to review all MDC safety evaluations. (Severity Level IV, 8326)
- The licensee failed to provide for an independent emergency preparedness program review. (Severity Level V, 8327)
- "Rejected Tags" were not being used to identify materials that were rejected and awaiting disposition. (Severity Level V, 8409)
- Failure to perform a safety review of a change made to the facility; e.g., the licensee was maintaining only one portable gasoline powered pump onsite instead of two pumps required by the Updated Safety Analysis Report (USAR). (Severity Level IV, 8411)
- Unreviewed safety question applicable to the licensee's acceptance of Standby Gas Treatment (SGT) System flow rate less than the design flow rate. (Severity Level V, 8420)
- The licensee accepted, without identifying and documenting on the Receipt Inspection Report, discrepancies in furnished ultrasonic examination and solution annealing heat treatment records. (Severity Level IV, 8501)
- The licensee failed to upgrade station drawings affected by MDCs 83-30 and 84-100. (Severity Level IV, 8501)

The following deviation was attributed to activities in this functional area:

- The licensee failed to issue a revision to Quality Assurance Instruction QAI-7, "Quality Records Retention Storage and Disposition," prior to August 31, 1983, as committed. (8326)

The following LER was attributed to activities in this functional area:

- When in calibrate mode, the Average Power Range Monitor (APRM) flow biased scram and rod block flow signals are full flow values, whereas the Technical Specification states that zero flow signals will be present. (LER 84-09)

During this assessment period, major management changes were implemented at the CNS and NPPD general offices. The nuclear operations division structure was trimmed and restructured so that the plant manager (division manager of nuclear operations) reports to the NPPD general manager through the assistant general manager - nuclear. The permanent plant staff level increased from approximately 150 to 250 people. Other improvements included increased communications between the site and corporate managements, increase of salaries in selected departments, changes to CNS departmental structure, increase of SORC and SRAB activities, establishment of an onsite compliance/commitment tracking system, training, emergency planning, quality assurance, and retention of experienced site engineering and operations departments personnel.

The previously listed violations and LERs indicate that there are still recurring problems in the areas of SRAB, SORC, QA, design change program, material controls, 10 CFR 50.59 reviews, procedure controls and adequacy, records storage, and the CNS Technical Specifications.

The previous SALP identified weaknesses. There have been changes and in some instances improvements. Improvements include the installation of the emergency notification system, upgraded operator morale, increased promotion opportunity onsite, the quality of site leadership, and implementation of some of the management appraisal recommendations.

There is still a significant deficiency in the nuclear expertise available above the site level. Upgrading of site titles has not changed the experience level in the district offices. There is an apparent tendency not to match authority to act with line responsibility. Additionally, line responsibility is often clouded by the tendency to manage through committee and an apparent lack of recognition that, in regard to matters affecting nuclear safety, the license conditions, Technical Specifications, regulations, and licensee commitments are applicable both onsite and at any location where the safety-related activity is carried out by district or contracted employees.

2. Conclusions

There has been improvement in management control. This is most evident onsite. Although there have been improvements, the overall effectiveness of management control is still questionable in some areas. Primarily because of onsite improvements, this functional area is considered to be Category 2.

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in the areas of quality assurance and administrative controls should remain at normal levels for onsite activities but should be increased for offsite support areas.

b. Recommended Licensee Actions

Licensee management attention should be directed toward assuring that not only the specific recommendations of the management appraisal are implemented but that the overall implementation effectively captures the intent of these recommendations.

Management support of the site should include adequate resources, training, and technical support. Specific emphasis is needed in programmatic controls as noted in the PAT inspection report and in the quality control for suppliers. Moreover, increased management attention should be directed toward assuring that CNS adequately implements the evolving industry standards and regulatory requirements which safety has dictated.

V. SUPPORTING DATA AND SUMMARIES

A. Licensee Activities

1. Major Outages

See Section IV.H.

2. Power limitations

The reactor was not limited in power level below the licensed limits during this appraisal period.

3. License Amendments

- No. 84, issued January 20, 1984, "Inservice Inspection Program."
- No. 85, issued March 20, 1984, "Corporate and Site Organization Changes."

- No. 86, issued June 1, 1984, "Licensee Event Reporting."
- No. 87, issued September 19, 1984, "Incorporation of Generic Letter 83-02/NUREG-0737 Requirements."
- No. 88, issued September 11, 1984, "Standby Gas Treatment Requirements During Refueling."
- No. 89, issued December 24, 1984, "Radiological Effluent Technical Specifications."
- No. 90, issued January 31, 1985, "Incorporation of Generic Letter 83-36/NUREG-0737 Requirements."

4. Significant Modifications

Major modifications completed during this appraisal period include construction of the new multipurpose facility (MPF), enlargement of the security building, addition of a new 700,000 gallon condensate storage tank, completion of the new meteorological tower, and installation of a new security computer. Many modifications are in progress such as installation of a new plant computer, SPDS, PMIS, installation of reactor recirculation system piping, GE Type 120 VAC HFA relays replacement, ADS logic, replacement of the turbine DEH computer, and other modifications resulting from 10 CFR Part 50, Appendix R, and environmental qualification of electrical equipment commitments.

5. Implementation of the CNS Emergency Plan

The CNS Emergency Plan was activated twice during this evaluation period. The emergency conditions that occurred are as follows:

- A Notification of Unusual Event was declared on April 19, 1984, due to the simultaneous loss of both trains of the Standby Gas Treatment System. The event was terminated on April 19, 1984, when the reactor reached the COLD SHUTDOWN condition.
- A Notification of Unusual Event was declared on June 15, 1984, because the river level adjacent to the plant reached 895 feet above mean sea level (MSL) and potential site flooding conditions existed. The river level subsequently crested at 897' 11" MSL. The event was terminated on June 21, 1984, when river level decreased to less than 897 feet above MSL. Site grade is 902 feet above MSL and shutdown is mandatory at a river level of 902 feet above MSL.

B. Inspection Activities

1. Violations

See Table 1

2. Major Inspections

The three special inspections listed below were conducted during this appraisal period:

- An inspection was conducted on August 1-2, 1984, of the licensee's proposed radiation protection program for the removal of existing reactor recirculation system piping and installation of new pipe. (Report 50-298/84-16)
- An inspection was conducted on October 22 through November 2 and November 13 through 16, 1984, by the NRC PAT team. The inspection included the licensee's management controls in the areas of plant operations, training, QA, radiological controls, procurement, maintenance, design changes, corrective action systems, and committee activities. Within the 9 areas inspected, 29 potential enforcement findings were identified. The findings, referred to in the inspection report as unresolved items, will be followed up by the NRC Region IV office during the subsequent SALP period. The findings are included in the analysis sections of this report as appropriate. (Report 50-298/84-21)
- An inspection was conducted on November 13-16, 1984, in the area of Technical Specification required surveillance testing of 125 volt and 250 volt station batteries. (Report 50-298/84-26)

C. Investigations and Allegations Review

None.

D. Escalated Enforcement Actions

1. Civil Penalties

One notice of violation with a proposed imposition of civil penalty was issued by the NRC to the licensee for violations occurring during this assessment period. The violations were identified in November 1984 and involved a failure of the licensee to demonstrate operability of station batteries and to have procedures for conducting battery charges. The violations were evaluated in the aggregate at Severity Level III and the NRC proposed the imposition of a civil penalty in the amount of $25,000 after the end of this assessment period.

2. Orders

No orders were issued relating to enforcement.

E. Management Conferences

1. Conferences

The following conferences were held during this appraisal period:

- An enforcement conference was conducted at the CNS with licensee management on October 4, 1983. The conference was held for the purpose of discussing the facts and circumstances relating to an apparent violation of the CNS Technical Specification requirements applicable to core alterations.
- Two meetings were conducted at the NRC Region IV office with licensee management on May 11 and May 21, 1984. The meetings were held to discuss the results of the March replacement and the May requalification examinations. Additional discussions were held in the latter meeting concerning the licensee's plans for increasing the number of training staff instructors, developing a formal training plan, and revising their emergency and abnormal operating procedures in order to clarify the distinction between immediate and followup operator actions.
- A meeting was conducted in the NRC Region IV office with licensee management on September 25, 1984. The meeting was held for the purpose of discussing licensee performance in the area of safety-related surveillance test performance.
- An enforcement conference was conducted at the CNS with licensee management on November 16, 1984. The conference was held to discuss an apparent violation of the CNS Technical Specification requirements applicable to testing station batteries.

2. Confirmation of Action Letters

None

F. Review of Licensee Event Reports and 10 CFR Part 21 Reports Submitted by the Licensee

1. Licensee Event Reports (LERs)

The SALP board reviewed the LERs submitted by the Nebraska Public Power District for the period of July 1, 1983, through January 31, 1985. This review included the following LERs:

- 83-13 through 83-15
- 84-01 through 84-11
- No LERs were issued during January 1985.

The NRC Office for Analysis and Evaluation of Operational Data (AEOD) performed reviews of licensee LERs, concentrating on the technical accuracy, completeness, and clarity of the event reports. Refer to Attachment 2 for details of those reviews.

2. Part 21 Reports

None

G. NRR Activities

1. NRR Site Visits

June 11 - 12, 1984, SPDS Pre-Implementation Audit
November 27 - 29, 1984, DCRDR Audit

2. Commission Briefing

None

3. Schedular Extensions Granted

October 3, 1983 - Equipment Qualification
July 2, 1984 - Fire Protection

4. Reliefs Granted

None

5. Exemptions Granted

September 21, 1983 - Fire Protection

6. License Amendments Issued

See Section V.A.3.

7. Emergency Technical Specifications Issued

None

8. Orders Issued

Order Confirming Emergency Response Capability, February 22, 1984

Order Confirming Licensee Commitments on Pipe Crack Issues, September 1, 1983

9. NRR/Licensee Management Conferences

None

TABLE 1

INSPECTION ACTIVITY AND ENFORCEMENT

FUNCTIONAL AREA	NO OF VIOLATIONS IN EACH SEVERITY LEVEL V	IV	III	II	I
A. Plant Operations		3			
B. Radiological Controls	1	6	1		
C. Maintenance		1			
D. Surveillance			1		
E. Fire Protection	3	1			
F. Emergency Preparedness					
G. Security Safeguards	1	1			
H. Refueling/Major Outages		1			
I. Training					
J. Licensing Activities					
K. Quality Programs and Administrative Controls Affecting Quality	4	5			
TOTAL	9	18	2	0	0

ATTACHMENT 1

Docket No. 50-298

FACILITY: Cooper Nuclear Station

LICENSEE: Nebraska Public Power District

EVALUATION PERIOD: July 1, 1983 to January 31, 1985

PROJECT MANAGER: Ernest D. Sylvester

1.0 INTRODUCTION

This report contains NRR's input to the SALP review for the Cooper Nuclear Station. The assessment of the licensee's performance was conducted according to NRR Office Letter No. 44, NRR Inputs to SALP Process, dated January 3, 1984. This office letter incorporates NRC Manual Chapter 0516, Systematic Assessment of Licensee Performance.

2.0 SUMMARY

NRC Manual Chapter 0516 specifies that each functional area evaluated will be assigned a performance category (Category 1, 2 or 3) based on a composite of a number of attributes. The performance of the Nebraska Public Power District in the functional area of Licensing Activities is rated Category 1.

3.0 CRITERIA

The evaluation criteria used in this assessment are given in NRC Manual Chapter 0516 Appendix, Table 1, Evaluation Criteria with Attributes for Assessment of Licensee Performance.

4.0 METHODOLOGY

This evaluation represents the integrated inputs of the Operating Reactor Project Manager (ORPM) and those technical reviewers who expended significant amounts of effort on the Cooper Nuclear Station licensing actions during the current rating period. Using the guidelines of NRC Manual Chapter 0516, the ORPM and each reviewer applied specific evaluation criteria to the relevant licensee performance attributes, as delineated in Chapter 0516, and assigned an overall rating category (1, 2 or 3) to each attribute. The reviewers included this information as part of each Safety Evaluation Report transmitted to the Division of Licensing. The ORPM, after reviewing the inputs of the technical reviewers, combined this information with his own assessment of licensee performance and, using appropriate weighting factors, arrived at a composite rating for the licensee. This rating also reflected the comments of the NRR Senior Executive assigned to the Cooper SALP assessment. A written evaluation was then prepared by the ORPM and circulated to NRR management for comments, which were incorporated in the final draft.

The basis for this appraisal was the licensee's performance in support of licensing actions that were either completed or had a significant level of activity during the current rating period. These actions, consisting of amendment requests, exemption requests, responses to generic letters, TMI items, and other actions, are classified as follows:

- 19 Multi-Plant Actions (8 completed): included in this category are:

  Control of Heavy Loads (C-10) - completed
  Appendix I Tech Spec Implementation Review (A-02) - completed
  Environmental Qualification of Safety-Related Electric Equipment (B-60) - completed
  Long Term Review - Containment Purge and Vent (B-24)
  Inspection of Stainless Steel Piping, Generic Letter 84-11 (B-84)

- 22 TMI (NUREG-0737 and Supplement 1) Actions (12 completed): included in this category are:

  Item II.K.3.28, Qualification of ADS Accumulators
  Item I.D.1, Control Room Design Review
  Item I.D.2, Safety Parameter Display System
  NUREG-0737 Tech Specs - completed
  NUREG-0737 Tech Specs, Generic Letter 83-36 - completed

- 26 Plant Specific Items (12 completed): included in this category are:

  Appendix R - Fire Protection of Safe Shutdown Capability - completed
  Mark I Containment Modifications - Tech Specs
  Hydrogen Recombiner Capability, Generic Letter 84-09
  Recirculation Piping Replacement Radiation Protection Program
  Snubber (Generic Letter 84-13) and Recirculation Bypass Tech Specs

5.0 ASSESSMENT OF PERFORMANCE ATTRIBUTES

The licensee's performance evaluation is based on a consideration of three of the seven attributes specified in NRC Manual Chapter 0516. These are:

- Management Involvement and Control in Assuring Quality
- Approach to Resolution of Technical Issues from a Safety Standpoint
- Responsiveness to NRC Initiatives

For the remaining four attributes (enforcement, reportable events, staffing and training, and qualification effectiveness), no basis exists for an NRR evaluation for the functional area of Licensing Activities.

A. Management Involvement and Control in Assuring Quality

During the present rating period, Nebraska Public Power District (NPPD) management personnel involvement and attention to quality in issues of major safety significance was apparent especially with regard to recirculation system piping repair and replacement and resolution of NUREG-0737, Supplement 1 items. For large, complex issues such as these, a licensee task manager is assigned to coordinate and control the engineering tasks and assure timely completion of the activity. NPPD has a small licensing and engineering staff so most of the Cooper Nuclear Station (CNS) engineering work associated with licensing is contracted out. Therefore, a significant amount of managerial talent is devoted to contract administration. The success of contractor programs in most cases indicates close management involvement. There is evidence of systematic planning whereby management strives to anticipate problems and to schedule priorities in an organized manner. As discussed in Section C below, the licensee uses a computerized tracking system to monitor licensing action schedules. NPPD management involvement has been apparent in candid constructive discussions with NRC staff on complex licensing issues and in the cooperative resolution of routine items.

One area where management attention could be increased is in assuring that the technical content of licensing submittals continues to improve.

Based on the above discussion, an overall rating of 1 is assigned to the attribute.

B. Approach to Resolution of Technical Issues from a Safety Standpoint

NPPD management and staff personnel have generally demonstrated technical understanding of issues involving licensing actions. The licensee has utilized contractor service, where necessary, to assure adequate technical expertise is brought to bear on the resolution of technical issues related to licensing actions. NPPD consistently exhibits conservatism in the evaluation and resolution of technical issues related to significant safety matters. This has been demonstrated by the extensive program to implement a safety parameter display system (SPDS) and the attention given the repair and replacement of recirculation system piping. These programs have also demonstrated the licensee's ability to develop technically sound resolutions to complex safety issues. The licensee has, with contractor assistance, adequately supported the proposed changes resulting from the evaluation of safety concerns. In most cases the licensee has competently responded to staff requests for information.

One area where licensee performance could be improved is in the technical justification accompanying applications for license amendments. In some cases this information has not been adequate to support the proposed amendment, necessitating further requests for information and clarification.

On the basis of these observations, a rating of 2 is assigned to this attribute.

C. Responsiveness to NRC Initiatives

NPPD has been responsive to the staff's requests for information and cooperative in complying with the staff's requirements. Use of a computerized tracking system for licensing actions has resulted in few items outstanding for significant periods of time. The licensee considers deadlines seriously and notifies NRR when they cannot be met. The licensee usually submits license amendment applications on a timely basis, allowing adequate time for NRR review. There has been no need for emergency changes to the Technical Specifications during the reporting period due, largely, to the licensee's activities to schedule all major projects and attempts to anticipate problems and minimize crises.

In meetings with NRR staff during this reporting period, the licensee was well prepared, responsive and made a concerted effort to resolve the issues. This was particularly evident at meetings with NRR involving the resolution of the equipment qualification program and the SPDS pre-implementation audit.

Based on the above considerations, an overall rating of 1 is assigned to this attribute.

D. Enforcement

No basis exists for an NRR evaluation of this attribute.

E. Reportable Events

No basis exists for an NRR evaluation of this attribute.

F. Staffing

Sufficient basis is not available for rating this attribute. However, NRR believes that a possible understaffing of the licensee's engineering support groups may exist and may contribute to the shortcomings in the content of some applications for license amendments.

G. Training and Qualification Effectiveness

No basis exists for an NRR evaluation of this attribute.

6.0 CONCLUSION

A complete performance rating of 1 has been assigned by NRR SALP evaluation effort for the current rating period.

Attachment 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 19 1985

MEMORANDUM FOR: John P. Jaudon, Chief
Project Section A
Reactor Project Branch 1
Region IV

FROM: Karl V. Seyfrit, Chief
Reactor Operations Analysis Branch
Office for Analysis and Evaluation
of Operational Data

SUBJECT: EVALUATION OF COOPER NUCLEAR STATION LICENSEE EVENT REPORTS FOR THE PERIOD JULY 1, 1983 TO JANUARY 31, 1985

The Office for Analysis and Evaluation of Operational Data has assessed the Licensee Event Reports (LERs) submitted under Docket No. 50-298 during the subject period. This has been done in support of the ongoing Systematic Assessment of Licensee Performance (SALP) review of the Nebraska Public Power District, with regard to their performance as licensee of the Cooper Nuclear Station. Our perspective would be indicative of that of a BWR system safety engineer who, although knowledgeable, is not intimately familiar with the detailed site-specific equipment arrangements and operations. Our review focused on the technical accuracy, completeness, and intelligibility of the LERs. Our review covered a majority of the LERs submitted during the assessment period.

The LERs reviewed were adequate in all important respects. The LERs typically provided clear descriptions of the cause and nature of the events as well as adequate explanations of the effects on both system function and public safety. Supplemental information was provided in three of the LERs reviewed. In most cases, the described corrective actions taken or planned by the licensee were considered to be commensurate with the nature, seriousness and frequency of the problems found. The enclosure provides additional observations from our review of the LERs.

In summary, our review of the licensee's LERs indicates that in all cases the licensee provided adequate descriptions of the events. In general, none of the LERs we reviewed involved what we would consider to be an especially significant event or serious challenge to plant safety.

If you have any questions, please contact either myself or Peter Lam of my staff. Dr. Lam can be reached at (301) 492-4438.

Karl V. Seyfrit

Karl V. Seyfrit, Chief
Reactor Operations Analysis Branch
Office for Analysis and Evaluation
of Operational Data

Attachments:
As stated

cc w/attachment:
D. Dubois, Region IV
E. Sylvester, NRR

SALP REVIEW FOR COOPER NUCLEAR STATION

The licensee submitted 15 LERs in the assessment period from July 1, 1983 to January 31, 1985. We reviewed 14 of these 15 LERs.

The LER review covered the following subjects and the general instructions of NUREG-0161. This SALP review is presented with the topic reviewed followed by the comments on that topic.

1. Review of LER for Completeness

   a. Is the information sufficient to provide a good understanding of the event?

      We found the information in the narrative sections to be adequately informative regarding the description of events, their associated consequences, and the licensee's corrective actions.

   b. Were the LERs coded correctly?

      All coded entries appeared to be correct. The codes selected by the licensee agreed well with the narrative descriptions.

   c. Was supplementary information provided when needed?

      Supplementary information was provided to clarify and expand the information in 3 of the 14 LERs reviewed.

   d. Were follow-up reports promised and submitted?

      No follow-up report was promised.

   e. Were similar occurrences properly referenced?

      The licensee appropriately referenced similar occurrences. Specifically, the following similar events were reported:

LERs Reviewed	Similar Occurrences
83-013/03L-0	LERs 77-57, 79-09 80-12, 81-15, 82-15

2. Multiple event reporting in a single LER

   The licensee did not report any multiple events in a single LER for the LERs which had been reviewed.

3. Relationship between PNs and LERs

The region issued three PNs during this review period. One of these PNs, PN-84-006 issued on April 19, 1984, needed to be followed by a LER. The licensee submitted the required LER (84-007-00) on May 18, 1984.

In summary, our review indicates that, based on the stated criteria, the licensee provided adequate descriptions of reportable events during the assessment period. No significant deficiencies were found in the LERs reviewed.

In Reply Refer To:
Docket: 50-285/85-07

MAY 14 1985

Omaha Public Power District
ATTN: R. L. Andrews, Division Manager -
Nuclear Production
1623 Harney Street
Omaha, Nebraska 68102

Gentlemen:

This refers to the Systematic Assessment of Licensee Performance (SALP) Board Report of the Fort Calhoun Station. The SALP Board met on April 16, 1985, to evaluate the performance of the subject facility for the period September 1, 1983, through February 28, 1985. The performance analyses and resulting evaluations are documented in the enclosed SALP Board Report.

The performance of your facility was evaluated in the selected functional areas identified in Section IV of the enclosed SALP Board Report.

The overall performance of Omaha Public Power District at the Fort Calhoun Station indicates a high level of dedication to nuclear safety as well as a high degree of technical competence in diverse areas essential to the safe operation of the facility.

It is of significant merit that the facility received Category 1 ratings in the following functional areas: plant operations, radiological controls, maintenance, fire protection, surveillance, refueling, and licensing activities. Additionally, a general trend toward a more favorable rating in quality programs and administrative controls affecting quality when compared to the previous SALP evaluation is noted. The decline in the performance category rating for security and safeguards indicates a need for aggressive management attention. The repeat Category 3 rating for training is based on the conclusion that the majority of OPPD's plans to upgrade this area are still in the developmental stage.

Unless you specifically request it, a management meeting will not be scheduled for this reporting period. Any request for a management meeting should be made, in writing, within 10 days of receipt of this report.

Any comments you may have regarding our evaluation of the performance of your facility should be submitted to this office within 30 days of issue of the SALP report. We specifically request that you advise us of actions you plan to take in response to the Category 3 ratings in the security and safeguards, and training functional areas. Your comments and our disposition of the same will be issued as appendices to the SALP Board Report.

Comments which you may submit are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL-96-511.

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Sincerely,

Original signed by
Robert D. Martin

Robert. D. Martin
Regional Administrator

Enclosures:
Appendix - NRC SALP Board Report 50-285/85-07
  Attachment 1 - NRR Supplement
  Attachment 2 - AEOD Supplement

cc w/enclosures:

W. G. Gates, Manager
Fort Calhoun Station
P.O. Box 399
Fort Calhoun, Nebraska 68023

Harry H. Voigt, Esq.
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1333 New Hampshire Avenue, NW
Washington, DC 20036

Kansas State Dept. Health
Nebraska State Dept. Health

bcc to DMB (IE40)

bcc distrib. by RIV: (Cont. on next page)

*RPB1
*RPB2
*EP&RPB
*RIV File
*Resident Inspector
*Section Chief (RPB2/A)
R. P. Denise, DRS&P
R. D. Martin, RA
*MIS System
*RSTS Operator
*D. Weiss, LFMB (AR-2015)
All RRIs (9)

*w/766

M. E. Murphy, RIV
ALL RRIs
Records Center, INPO
J. M. Taylor, D/IE
Chmn. N. J. Palladino (MS: H-1144)
Comm. P. M. Roberts (MS: H-1149)
Comm. J. K. Asselstine (MS: H-1149)
Comm. F. M. Bernthal (MS: H-1149)
Comm. L. W. Zeck (MS: H-1149)

SALP BOARD REPORT

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

SYSTEMATIC APPRAISAL OF LICENSEE PERFORMANCE

50-285/85-07

Omaha Public Power District

Fort Calhoun Station

September 1, 1983 - February 28, 1985

I. INTRODUCTION

The Systematic Assessment of Licensee Performance (SALP) program is an integrated NRC staff effort to collect available observations and data on a periodic basis and to evaluate licensee performance based upon this information. SALP is supplemental to normal regulatory processes used to ensure compliance to NRC rules and regulations. SALP is intended to be sufficiently diagnostic to provide a rational basis for allocating NRC resources and to provide meaningful guidance to the licensee's management to promote quality and safety of plant operation.

An NRC SALP Board, composed of the staff members listed below, met on April 16, 1985, to review the collection of performance observations and data to assess the licensee performance in accordance with the guidance in NRC Manual Chapter 0516, "Systematic Assessment of Licensee Performance." A summary of the guidance and evaluation criteria is provided in Section II of this report.

This report is the SALP Board's assessment of the licensee's safety performance at Fort Calhoun Station for the period September 1, 1983, through February 28, 1985.

SALP Board for Fort Calhoun Station:

R. P. Denise, Director, Division of Reactor Safety and Projects (Chairman)
R. L. Bangart, Director, Division of Radiation Safety and Safeguards
L. E. Martin, Section Chief, Project Section A, Reactor Project Branch 2
L. A. Yandell, Senior Resident Inspector
J. R. Miller, Chief, Operating Reactors Branch 3
E. G. Tourigny, Project Manager

Attendees at all or part of the SALP Board Meeting were:

R. E. Hall, Chief, Emergency Preparedness and Radiological Protection Branch
W. C. Seidle, Technical Assistant
J. B. Baird, Chief, Emergency Preparedness Section
R. J. Everett, Chief, Nuclear Materials Safety Section
R. E. Baer, Regional Inspector

II. CRITERIA

Licensee performance was assessed in 11 selected functional areas. Each functional area normally represents areas significant to nuclear safety and the environment, and are normal programmatic areas.

One or more of the following evaluation criteria were used to assess each functional area.

1. Management involvement and control in assuring quality.
2. Approach to resolution of technical issues from a safety standpoint.
3. Responsiveness to NRC initiatives.
4. Enforcement history.
5. Reporting and analysis of reportable events.
6. Staffing (including management).
7. Training effectiveness and qualification.

However, the SALP Board is not limited to these criteria and others may have been used where appropriate.

Based upon the SALP Board assessment each functional area evaluated is classified into one of three performance categories. The definition of these performance categories is:

Category 1. Reduced NRC attention may be appropriate. Licensee management attention and involvement are aggressive and oriented toward nuclear safety; licensee resources are ample and effectively used so that a high level of performance with respect to operational safety is being achieved.

Category 2. NRC attention should be maintained at normal levels. Licensee management attention and involvement are evident and are concerned with nuclear safety; licensee resources are adequate and are reasonably effective so that satisfactory performance with respect to operational safety is being achieved.

Category 3. Both NRC and licensee attention should be increased. Licensee management attention or involvement is acceptable and considers nuclear safety, but weaknesses are evident; licensee resources appear to be strained or not effectively used so that minimally satisfactory performance with respect to operational safety is being achieved.

The SALP Board has also categorized the performance trend over the course of the SALP assessment period. The trend is meant to describe the general or prevailing tendency (the performance gradient) during the SALP period. This categorization is not a comparison between the current and previous SALP ratings; rather the categorization process involves a review of performance during the current SALP period and categorization of the trend of performance during that period only. The performance trends are defined as follows:

Improved: Licensee performance has generally improved over the course of the SALP assessment period.

Same: Licensee performance has remained essentially constant over the course of the SALP assessment period.

Declined: Licensee performance has generally declined over the course of the SALP assessment period.

III. SUMMARY OF RESULTS

In summary, the licensee has exhibited significant strength in the areas of plant operations, radiological controls, maintenance, fire protection, surveillance, refueling, and licensing activities. Weak areas identified included security and safeguards, and training. The licensee's performance and trend are summarized in the table below along with the performance category from the previous SALP evaluation period:

	Functional Area	Previous Performance Category (9/1/82 to 8/31/83)	Present Performance Category (9/1/83 to 2/28/85)	Trend During Latest SALP Period
A.	Plant Operations	1	1	Declined
B.	Radiological Controls		1	Improved
	1. Radiation Protection	2	N/A	
	2. Chemistry/Radiochemistry and Confirmatory Measurements	3	N/A	
	3. Radwaste Management, Effluent Releases, and Effluent Monitoring	2	N/A	
	4. Transportation/Solid Radwaste	3	N/A	
	5. Environmental Monitoring	2	N/A	
C.	Maintenance	2	1	Improved
D.	Surveillance	1	1	Same
E.	Fire Protection	2	1	Same

F.	Emergency Preparedness	2	2	Improved
G.	Security and Safeguards	2	3	Improved
H.	Refueling	1	1	Same
I.	Quality Programs and Administrative Controls Affecting Quality*	3	2	Improved
J.	Licensing Activities	2	1	Improved
K.	Training	3	3	Improved

*This category was divided into two individual categories in the previous SALP report.

The total NRC inspection effort during this SALP evaluation period consisted of 44 inspections including resident inspector inspections and emergency exercises for a total of 3,241 direct inspection man-hours.

IV. PERFORMANCE ANALYSIS

A. Plant Operations

1. Analysis

This area was inspected on a continuing basis by the NRC resident inspector. Four violations and no deviations were identified in this functional area during the appraisal period:

- Failure to properly terminate a containment pressure reduction when the limiting X/Q was exceeded. (Severity Level IV, 8407-01)
- Failure to provide an adequate procedure requiring independent verification of tag-outs to ensure that equipment was properly isolated. (Severity Level IV, 8412-01)
- Failure of shift supervisor to document review of maintenance orders. (Severity Level IV, 8412-02)
- Failure to install the locking device on a closed valve as required by the procedure lineup. (Severity Level IV, 8429-02)

The four Licensee Event Reports (LERs) listed below involved activities in the area of plant operations:

- Inadvertent opening of breaker supplying power to a control room DC panel. (LER 84-003)
- Unplanned actuation of VIAS while shifting to the high alert/alarm setpoints. (LERs 84-014, 84-019, and 84-024)

Fort Calhoun Station maintains an experienced group of senior operators and reactor operators. The plant manager, three of the five technical area supervisors, three training supervisors, and the plant engineer all hold and maintain senior reactor operator licenses and provide support and technical expertise to the operations department. Operating personnel exhibit a strong commitment to procedural compliance and a good understanding of technical issues associated with plant operations. These strengths have been observed by NRC inspectors during emergency preparedness drills, routine operations, and abnormal plant situations. The Fort Calhoun Station completed 302 days of continuous operation (the longest in the plant's history) during this evaluation period which is indicative of the high caliber of onshift personnel. The plant experienced only one reactor trip this SALP period, the first since December 1982.

The operations department has experienced a net loss of licensed personnel during this appraisal period. Attrition due to resignations, retirement, and one transfer to another department prevents the licensee from manning a full six-shift rotation as they did on January 1, 1984, when the new manning regulations went into effect. Additional burden was placed on the licensed operators due to long term disabilities, hospitalizations, and the removal of one senior operator from the shift rotation for training following his failure of the annual requalification examination. A review of licensee statistics for the last quarter of this appraisal period indicated that overtime work hours are increasing.

The licensee continued their program of upgrading the Fort Calhoun Station annunciator system to eliminate nuisance alarms and to have no annunciators lit during power operation. Significant progress was made during the last refueling outage with work continuing in this area as part of the overall control room design review being done in accordance with Supplement 1 to NUREG-0737 (Generic Letter 82-33), Item 1.D.1. Portions of the safety parameter display system were energized this report period and are available for use by the operators.

2. Conclusions

The overall performance level of the operations department has been excellent during this appraisal period. The net loss of licensed personnel and the lack of qualified replacements are a concern to the NRC.

The licensee is considered to be in Performance Category 1 in this area.

Trend: Declined

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area could be reduced.

b. Recommended Licensee Actions

Licensee management should give priority to reversing the declining trend related to staffing and staff qualifications to ensure that the licensee meets Fort Calhoun Station staffing objectives. Efforts to improve the control room annunciator system and to implement the recommendations of the control room design review should be continued.

B. Radiological Controls

1. Analysis

Eight inspections were conducted during the assessment period by region-based inspectors concerning radiological controls. These eight inspections covered the following areas: radiation protection-normal operations; radiation protection-refueling outage; radwaste management, effluent releases, and effluent monitoring; chemistry/radiochemistry and confirmatory measurements; and transportation/solid radwaste. One violation and no deviations were identified:

. Failure to Follow Procedures. (Severity Level V, 83-31)

a. Radiation Protection

This area was inspected twice during normal plant operations and once during a refueling outage.

The person-rem for 1983 was 433 as compared to the PWR national average of 592. In 1984, 544 person-rem was expended. The 1984 national averages have not been tabulated, but the 1984 values are expected to be near the 1983 levels. The licensee's person-rem average between 1978 and 1983 was 385 which is below the PWR national average of 556 for the same time period.

The radiation protection staff was stable during the assessment period. All members of the radiation protection staff met the ANSI N18.1-1971 qualification requirements as senior radiation protection technicians. The radiation protection staff consisted of 12 technicians, 2 supervisors, and 2 ALARA coordinators. The licensee had decreased reliance on contractor technicians; only one contractor technician was included in the radiation protection staff.

The lack of a full-time ALARA coordinator was noted as a concern in the previous assessment. In this assessment period, the licensee established and filled an ALARA coordinator and an assistant ALARA coordinator position. The implementation of an extensive decontamination program in the auxiliary building corridors, that allowed access into these areas without the need for protective clothing, was noted as a program improvement in the previous assessment. The licensee had continued to expand this decontamination effort to include additional areas and rooms in the auxiliary building. The lack of timely action to resolve open items was identified as a major concern in the previous assessment. In this assessment period, good progress had been made toward the resolution of outstanding open items.

A comprehensive training program has not been implemented for the radiation protection staff.

b. Chemistry/Radiochemistry and Confirmatory Measurements

This area was inspected once during the assessment period which included onsite confirmatory measurements with the Region IV mobile laboratory. The following LERs concerning this area were submitted:

- Boron concentrations in the Safety Injection and Refueling Water Tank below Technical Specification limits. (LERs 84-012 and 84-021)

An overall improvement was noted in the percent agreement between the NRC's and the licensee's measurements in this assessment period. Considerable improvement was made in the percent agreement concerning the comparative analyses of gaseous samples. The percent agreement for gaseous samples in this assessment was about 70 percent as compared to less than 50 percent in the previous assessment period. The licensee had implemented a quality control cross-check program with an offsite independent laboratory.

An approximate 60 percent turnover rate was noted among the chemistry/radiochemistry staff in the assessment period.

A comprehensive training program had not been implemented for radiochemistry personnel.

c. Radwaste Management, Effluent Releases, and Effluent Monitoring

This area was inspected once in the assessment period. The following LERs concerning this area were submitted:

- Exceeded Technical Specification Limits for I-131 Dose Equivalent in Reactor Coolant. (LER 84-004)
- Unplanned actuation of Ventilation Isolation Actuation Signal (VIAS) due to instrument error. (LERs 84-005, 006, 007, 010, 017, 018, and 023)
- Disconnection of sample line from auxiliary building ventilation duct to gaseous effluent monitor. (LER 84-011)
- Gaseous leak in the waste gas vent header caused stack iodine monitor to initiate VIAS. (LER 84-025)

The licensee had a well established program for sampling and analyses of liquid and gaseous samples to assure compliance with Technical Specification requirements. Improvements noted in this area included: installation of low-background steam generator blowdown radiation monitors; in-place calibration of liquid and gaseous effluent monitors with liquid and gas standards; and an in-place testing program for the auxiliary building HEPA filters.

Approximately 20 unplanned actuations of the VIAS related to instrument error were reported in LERs during the assessment period.

d. Transportation/Solid Radwaste

This area was inspected twice during the assessment period. The licensee had updated their program to include the revisions to Department of Transportation regulations effective July 1, 1983, and revisions to 10 CFR Part 20.311, 10 CFR Part 61.55, and 10 CFR Part 61.56 effective December 27, 1983. Improvements noted in this area included: assignment of a full-time radwaste coordinator; development of detailed shipping procedures and records; implementation of a QA/QC program concerning the packaging of radioactive materials; and establishing a procedure to evaluate the amount of radwaste generated.

A comprehensive training program had not been established for personnel involved with transportation/solid radwaste activities.

e. Environmental Monitoring

The radiological environmental monitoring program was inspected once during the assessment period. No significant problems were identified. There had been no turnover in the onsite staff responsible for the program during the past several years. Improvements noted in this area included the use of persons with expertise in environmental monitoring on the team responsible for auditing the radiological environmental monitoring program and implementation of comprehensive procedures for identification, collection, and shipping of environmental samples.

A formal training program had not been established for persons involved with the radiological environmental monitoring program.

2. Conclusions

Improvements were noted in the area of radiological controls in this assessment period as compared to the previous assessment and these improvements reflect strong management attention to weaknesses previously identified. The licensee had made good progress toward resolution of outstanding open items. During the assessment period, 31 open items were resolved and only one new open item was identified. A high turnover rate was noted among the chemistry/radiochemistry staff. Comprehensive training programs had not been established that included schedules, goals and objectives, full-time instructors, lesson plans, and training aids.

No problems were noted concerning enforcement history, resolution of technical issues, management involvement, and responsiveness to NRC initiatives.

The licensee is considered to be in Performance Category 1 in this area.

Trend: Improved

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area could be reduced.

b. Recommended Licensee Actions

Management attention is needed to ensure that comprehensive training programs are established. Management should investigate the cause of the high turnover rate in the chemistry/radiochemistry staff and establish any appropriate remedial actions.

C. Maintenance

1. Analysis

This area was inspected by region-based NRC inspectors and on a continuing basis by the NRC resident inspector. Four violations and no deviations were identified in this functional area during the appraisal period:

- Failure to have proper receipt inspection of material performed prior to installation by maintenance personnel. (Severity Level V, 8415-01)
- Failure by maintenance craftsman to properly verify and use the correct revision of a surveillance test. (Severity Level IV, 8418-02)
- Failure to provide adequate procedures:
  a. to identify and resolve nonconformances associated with plant process instrumentation, surveillance test

instrumentation, and pressure gauges when found to be out-of-tolerance at calibration; and

b. to implement the cleaning requirements for fluid systems and associated components in accordance with ANSI N45.2.1-1973. (Severity Level V, 8421-01)

- Failure to follow procedures:

  a. by failing to perform investigation to determine the effect of secondary standards being out-of-tolerance; and

  b. by failing to provide minimum calibration schedules for oscilloscopes and electrical current measuring standards. (Severity Level IV, 8421-02)

The twelve LERs listed below involved activities in the area of maintenance:

- Diesel generator field failed to flash during surveillance test. (LERs 83-008 and 83-011)
- Reactor Coolant Loop RC-2A flow indicator loop failed high. (LER 83-009)
- Low pressure safety injection pump sequencer timer failed during monthly ESF surveillance test. (LER 83-010)
- Failed power supply fuse in RPS channels. (LER 83-012)
- Containment pressure switches were found out-of-tolerance during surveillance test. (LER 83-013)
- Main steam safety valves failed to lift within setpoint values during surveillance test. (LER 84-002)
- Steam Generator RC-2B tube failure. (LER 84-008)
- Electrical penetration assemblies failed under environmental qualification testing being conducted by the test laboratory to fulfill the requirements of 10 CFR Part 50.49. (LER 84-009)
- Noise spikes caused tripping of the "A" and "C" Thermal Margin Low Pressure reactor protective system trip circuits which resulted in a reactor trip. (LER 84-013)

- High alarm setpoint for stack noble gas monitor was found out-of-tolerance during surveillance test. (LER 84-016)

- Hydrogen analyzers failed to indicate proper concentration during calibration. (LER 84-020)

The major maintenance efforts accomplished this appraisal period occurred during the refueling outage March 5 to July 8, 1984. Maintenance activities accomplished included eddy current testing of both steam generators, sludge lancing of the secondary side of both steam generators, repair of the reactor coolant pump gaskets, work on the Quality Safety Parameters Display System (QSPDS), and replacement of the high pressure turbine rotor. Extensive management involvement at the planning level and expanded use of the computerized tracking system enabled these activities to be accomplished essentially on schedule.

Three additional, events caused an extension to the originally planned outage. The first was the removal of the drilled tube support plate rim in both steam generators to reduce the potential for tube "denting" caused by stresses in the support plate. This "rim cut" was accomplished by contract boilermakers under the direct supervision of OPPD personnel and completed within two weeks. The licensee utilized mockup training for all craftsmen, and this contributed to the timely completion of the job and to the less than anticipated overall man-rem exposure. The second event was the steam generator tube failure in RC-2B that occurred on May 16, 1984, during the reactor coolant system leak test. The licensee had identified a small tube leak prior to shutdown, and had expended considerable effort to identify the cause prior to startup. This effort included an extensive eddy current test program, helium leak checks before and after sludge lancing, and dye leak tests, but no positive indication of a leak was identified prior to the reactor coolant system leak test. Following the tube failure, the licensee removed the failed tube from RC-2B, performed eddy current testing on all accessible tubes on both steam generators, performed a lab analyses on the failed tube, initiated more restrictive limits on the primary to secondary leak rate, and instituted vendor recommended chemistry control procedures. The third event was precipitated when a testing laboratory informed OPPD that containment penetration lead wire insulation could fail under the harsh environment of a Large Break Loss of Coolant Accident (LBLOCA). To correct this problem, the licensee developed a cable splice using qualified sleeves that shielded the lead wire insulation from the harsh environment. A total of 638 splices involving 48 cables were reworked during this 2-week extension of the outage. With regard to all three of these events, the licensee management responded quickly to address these matters and to provide the necessary resources and support to these activities.

Another major maintenance effort undertaken by Fort Calhoun during this appraisal period was the installation of new spent fuel storage racks by an outside contractor. While removing the old racks, there were instances where plant QC personnel stopped work because of the improper way the job was being performed and controlled. During the removal of one spent fuel rack, the wire rope sling broke and the rack became wedged in the pool. OPPD failed initially to exercise appropriate management control by providing adequately trained people to direct this work and sufficient QC coverage to monitor contractor activities.

The licensee's system for tracing measuring and test equipment (M&TE) usage was found to be weak and lacked adequate procedures to identify all items inspected, tested, or measured to specific M&TE. Adequate procedures were not established to make disposition of possible nonconformances of those items when M&TE was found to be out-of-tolerance. The licensee revised plant procedures and expanded the M&TE program during this SALP period to address those concerns identified in NRC Inspection Reports 50-285/84-12 and 50-285/84-21.

The previous SALP identified that several long term electrical jumpers dating back to 1973 were still outstanding. The licensee made significant progress during this SALP period at closing out these items and incorporating them as permanent design changes in accordance with the plant standing order.

The tracking of maintenance orders (MOs) was another item of concern discussed in the previous SALP. A computerized MO system is now in effect that works in conjunction with a computerized tracking system to provide current status information.

The tracking and closeout of design changes was a third item of concern discussed in the previous SALP report. The efforts of the update team were completed this SALP period, and all modifications through December 31, 1980, have been reviewed. A computerized listing of all outstanding Design Change Requests and Engineering Evaluation and Assistance Requests was established and licensee management now has the capability of assessing the status of the design change program.

The licensee implemented a Qualified Life Program (QLP) during this appraisal period to "establish and maintain the qualified life of safety-related equipment installed in a harsh environment." Although the QLP was established in April 1984, the licensee was slow in fully implementing the program and delegating appropriate responsibility to the various crafts. A full-time coordinator was assigned to this effort in order to: (1) review Electrical Equipment Qualification (EEQ) documentation since the QLP started, (2) review QLP entries for accuracy in the computer data base used in the maintenance program, and (3) provide comprehensive training of the QLP/EEQ program.

2. Conclusions

The licensee has initiated additional management control programs to strengthen the maintenance area, and has established a better working interface between the plant and design groups at the Jones Street office. The resolution of items mentioned in previous SALP reports indicates OPPD's commitment in maintenance, but poor control of the new spent fuel rack installation and delays in full implementation of the EEQ program suggest areas that require additional management attention.

The plant has a stable, well-qualified maintenance staff that has seen little turnover the past three SALP periods.

The licensee is considered to be in Performance Category 1 in this area.

Trend: Improved

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area could be reduced.

b. Recommended Licensee Actions

The licensee should continue the increased management attention being given to the EEQ program in preparation for a site verification inspection by the NRC in 1985.

D. Surveillance

1. Analysis

This area was inspected on a continuing basis by the NRC resident inspector. No violations or deviations were identified in this area.

Fort Calhoun maintains a well-developed and effectively managed surveillance test program. A monthly surveillance testing schedule is published to ensure that all required tests are assigned as to due date and responsible department. The program is set up to verify that the current revision to the test is being used, that QC verification is obtained where required, and that calibrated test equipment is used and identified on the procedure. During the last refueling outage, the master schedule was updated and the applicable surveillance tests were written or modified to reflect new Technical Specification requirements.

2. Conclusions

The licensee maintains a well-developed and effectively managed surveillance test program.

The licensee is considered to be in Performance Category 1 in this area.

Trend: Same

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should remain at the reduced level.

b. Recommended Licensee Actions

The Board recommends that the licensee continue to exercise strong management control of the surveillance program.

E. Fire Protection

1. Analysis

This area was inspected on a continuing basis by the NRC resident inspector. No violations or deviations were identified in this area. The one LER listed below involved activities in the functional area of fire protection:

- Temporary fire barrier failed to meet design criteria. (LER 84-022)

A major activity in this area by OPPD during this appraisal period was the resolution of items identified by the NRC special inspection team and documented in NRC Inspection Report 50-285/83-12 dated July 1, 1983.

Another major effort by the licensee this report period was the upgrading of temporary fire barriers into permanent barriers and to grout conduit penetrations. This 3-month effort utilized a qualified outside contractor and resulted in approximately 350 fire barriers being upgraded. The licensee is presently developing a program to enable OPPD personnel to install new fire barrier penetrations and perform maintenance on permanent fire barriers.

2. Conclusions

The licensee has made significant progress at resolving the items identified in the fire protection audit report and is working toward compliance with the requirements of 10 CFR Part 50, Appendix R, Sections III.G and III.L.

The licensee is considered to be in Performance Category 1 in this area.

Trend: Same

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should remain at the present level.The NRC should complete the review and processing of OPPD's exemption requests.

b. Recommended Licensee Actions

The licensee management should continue their efforts to resolve all identified items pertaining to compliance with Appendix R.

F. Emergency Preparedness

1. Analysis

During the assessment period, five routine emergency preparedness inspections were conducted. Three of the inspections were routine reviews of the implementation status for various elements of the emergency preparedness program.

Two emergency exercise inspections were also conducted during the assessment period in conjunction with the licensee's annual emergency exercises held December 6-7, 1983, and October 24, 1984. No violations or deviations were observed by the NRC inspectors.

Most of the open items identified during inspections in the previous assessment periods were closed during this assessment period based on timely licensee actions. For concerns identified during the reporting period, corrective action was noted for most items during the next review of that item.

During the October 24, 1984, emergency exercise, the Federal Emergency Management Agency (FEMA) evaluated offsite emergency preparedness of states and local agencies. As a result of this evaluation, FEMA identified a Category A deficiency in regard to agreements for ambulance services in two counties of the state of Iowa side of the 10-mile emergency planning zone. Plans for resolution of this deficiency included agreements between Iowa and the ambulance services, training for ambulance personnel, and a drill to demonstrate this capability. The licensee has been active in this matter and is cooperating with state and local authorities to achieve resolution of this offsite preparedness concern.

Management involvement and control of the emergency preparedness program during the period appeared to be adequate for implementation of an effective program.

Staffing of the emergency preparedness program was also considered to be adequate during this reporting period. No reportable events in the emergency preparedness area were received during this reporting period.

2. Conclusions

The licensee has maintained an acceptable level of emergency preparedness during the period and demonstrated adequate capability to protect the health and safety of the public by conducting two successful emergency exercises. Management involvement and control has been adequate for implementation of an effective program. Responses to NRC concerns have been timely, thorough, and acceptable in most cases. Overall, the licensee's program appeared to have increased in effectiveness since the previous assessment period.

The licensee is considered to be in Performance Category 2 in this area.

Trend: Improved

3. Board Recommendations

a. Recommended NRC Action

The NRC inspection effort in this functional area should continue at a normal level.

b. Recommended Licensee Action

A more aggressive level of management attention to implementation of the emergency preparedness program should be pursued. Additional management attention should be given to resolution of each of the NRC identified concerns in a timely manner.

G. Security and Safeguards

1. Analysis

The physical security staff performed five inspections during this SALP period.

Twelve violations were identified in this functional area during the appraisal period.

- Failure to demonstrate that the microwave system is tested quarterly against the manufacturer's design specifications. (Severity Level IV, 8417-02)
- Failure to take proper compensatory action when the perimeter intrusion detection system coverage was discovered to be ineffective. (Severity Level IV, 8417-03)
- Failure to provide authorized escort. (Severity Level IV, 8418-01)
- Failure to report facility modification. (Severity Level *, 8420-01)
- Inadequate key control. (Severity Level *, 8420-02)
- Failure to provide an adequate barrier for part of one vital area. (Severity Level IV, 8420-03)
- Inadequate compensatory measures. (Severity Level *, 8420-04)
- Inadequate surveillance television coverage. (Severity Level *, 8420-05)
- Failure to provide adequate access control for a vital area. (Severity Level IV, 8420-06)
- Insufficient search at access control point. (Severity Level *, 8420-07)

- Failure to maintain operable assessment aids. (Severity Level *, 8420-08)

- Inadequate compensatory measures following failure of search equipment. (Severity Level *, 8420-09)

*Violations 8420-01, 02, 04, 05, 07, 08, and 09 were categorized individually at Severity Level IV or V, but were considered in the aggregate as a Severity Level III Violation.

Three deviations were identified in this functional area during the appraisal period.

- Failure to revise security plan according to commitment made to NRC. (8326-04)

- Failure to provide continuous monitoring. (8420-10)

- Failure to perform effectiveness test. (8420-11)

Forty-nine licensee event reports (LER) of physical security events were submitted in accordance with 10 CFR Part 73.71:

- Loss of primary (CPU1) and secondary (CPU2) security computers. (LER's Nos. 83-07 through 12, 84-02 through 27, 84-29 through 34, and 85-01 through 09)

- Loss of AC power and failure of uninterruptible power source to operate the security computers. (LER No. 84-01)

- Reduction in offsite communications to local law enforcement agencies. (LER No. 84-28)

The licensee lost use of his primary (CPU1) and secondary (CPU2) security computers 47 times during this SALP period. On one occasion, the licensee lost AC power and his backup source of uninterruptible power failed to operate automatically. In LER 84-28, the licensee lost "normal" offsite communications to the local area due to an accidental slicing of the underground cable by an offsite nonutility construction crew.

The licensee was issued a Confirmatory Action Letter (CAL) on August 16, 1984. The CAL was issued as a result of a Region IV inspection in July 1984. An enforcement conference was held October 11, 1984, and documented in NRC Inspection Report 50-285/84-27. The enforcement conference was called to discuss the apparent violations from the special security inspection (NRC Inspection Report 50-285/84-20) conducted during the period August 20-24, 1984. The importance of management involvement in establishing an effective security program was emphasized by the NRC participants. The need to develop and implement an effective testing and maintenance program for security-related equipment was discussed. As a consequence of the August special security inspection, a Notice of Violation and Proposed Imposition of Civil Penalty (CP) dated February 14, 1985, was issued to the licensee. The CP was reduced by 50 percent to $25,000 because of prior good performance in the area of concern; specifically, no previous escalated enforcement actions and repeated ratings of Category 2 in their (SALP) evaluations.

Subsequent to the August 20-24, 1984, inspection, OPPD initiated an extensive security improvement program. OPPD management approved a Fort Calhoun Station security organizational change providing for a dedicated security supervisor reporting to the Supervisor-Administrative Services and Security at Fort Calhoun Station, a review of the maintenance and testing program relating to security equipment, and a review of the entire physical security plan by a qualified security consultant. In addition, Fort Calhoun Station made a commitment to submit a revised physical security plan to the NRC by May 1, 1985.

2. Conclusions

The licensee had not placed emphasis and dedication on maintaining an effective security program and management involvement with security matters has been marginal. The result of management's lack of commitment to security was evidenced by insufficient training and maintenance, and by improper compensatory actions resulting in violations.

The licensee is considered to be in Performance Category 3 in this area.

Trend: Improved

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should be increased. An evaluation of the licensee's corrective actions and their impact on the security program should be performed by August 31, 1985.

b. Recommended Licensee Action

The level of licensee management attention evidenced during the last quarter of this SALP period should be continued.

The licensee should increase audit activity of security training, maintenance, and requirements of the security plan. Licensee management should ensure that security resources and organization are adequate to implement the security plan.

H. Refueling

1. Analysis

This area was inspected on a continuing basis by the NRC resident inspector during the period of refueling April 1-8, 1984. Fort Calhoun Station was in a refueling outage from March 5 to July 8, 1984, for Cycle 9 refueling. A total of 40 new bundles were inserted into the core.

No violations or deviations were identified during this evaluation period.

The licensee continued to utilize a fuel load/shuffle scheme designed to reduce neutron flux to the reactor vessel as part of OPPD's efforts to address the pressurized thermal shock issue. The fuel movement was completed without incident and the NRC inspector verified that Technical Specification requirements were satisfied.

2. Conclusions

The licensee management demonstrated excellent prior planning and effective control of refueling activities. The licensee is considered to be in Performance Category 1 in this area.

Trend: Same

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should remain at reduced levels.

b. Recommended Licensee Actions

Licensee management should continue its involvement in the planning of refueling outages, the observation of refueling activities, and adherence to procedures.

I. Quality Programs and Administrative Controls Affecting Quality

1. Analysis

This functional area was inspected on a continuing basis by the NRC resident inspector and by region-based NRC inspectors. Six violations and no deviations were identified in this area during the appraisal period:

- Failure to audit security procedures every 12 months. (Severity Level IV, 8326-01)
- Failure to adequately perform the review and approval steps of a CQE piping isometric as part of the post-installation modification review process. (Severity Level IV, 8335-01)
- Failure to provide procedures to assure that appropriate Fort Calhoun Station personnel were provided with current lists of CQE equipment. (Severity Level V, 8410-03)
- Failure to properly review OPPD's response to IE Bulletin 82-02 resulting in a material false statement being made that was contrary to actual practice at the Fort Calhoun Station. (Severity Level III, 8412-03)
- Failure to take prompt corrective action on the resolution of QA deficiency/quality reports within the required response period. (Severity Level IV, 8429-01)
- Failure to properly establish, monitor, and closeout temporary CQE storage areas. (Severity Level IV, 8501-01)

The two LERs listed below involved activities in the area of administrative controls:

- Auxiliary building crane interlocks were left in the bypass position without the crane supervisor being present. (84-001)
- A load of approximately 250 pounds was carried by the polar crane over the reactor coolant system when the fluid in the pressurizer was greater than 225 degrees F. (84-015)

The licensee submitted the revised OPPD QA Plan to the NRC for review on August 31, 1984. The program is fully implemented and the complete set of QA department procedures has been issued. These items were addressed in the last SALP period and the delay in implementing the new program was considered a weakness in the OPPD program.

It was determined during routine inspections by regional inspectors in the radiological controls area that there were no QA auditors on the licensee's onsite staff that had any training or background experience in radiation protection except for instrument calibration. The NRC determined during another inspection that the licensee did not have a program established to audit vendors that are contracted to perform radiochemical analyses on samples of Fort Calhoun Station radwaste for the requirements of 10 CFR Part 61.55. In the area of security and safeguards the NRC determined that the licensee had failed to perform an audit of security procedures and practices every 12 months as required by the Fort Calhoun Station Physical Security Plan.

During this appraisal period, major management changes were implemented at the OPPD corporate level and at the Fort Calhoun Station. A Nuclear Production Division was formed and a separate division manager for nuclear matters was established directly under the assistant general manager. The reorganization provided increased management attention to the operation of the Fort Calhoun Station. The NRC resident inspector observed that the manager of the Nuclear Production Division initiated weekly staff meetings at the site, made routine tours of the plant, and attended all NRC exit interviews at the site.

The licensee reorganized the site management structure to have the Quality Control (QC) section report to the Supervisor-Technical, instead of the Supervisor-Maintenance. Observations by the resident inspector indicated that this move provided a greater degree of independence for the QC section from the Maintenance Department and enhanced their effectiveness onsite. A Supervisor-Station Training position reporting directly to the plant manager was established this appraisal period and filled in February 1985. This reflects the licensee's commitment to provide increased management attention to plant training activities.

An area of continued long standing concern to the NRC is the matter of the Fort Calhoun Station construction QA records. An enforcement conference between OPPD and NRC personnel was held in the Region IV offices on September 9, 1983, to address this matter and identify the licensee's corrective actions. OPPD agreed to review "all commitments to the NRC for retention of design, procurement, manufacturing, installation and construction records, and all applicable involved codes, standards, and specifications." The licensee has examined the available records and identified those which were missing, along with an evaluation of their significance. A recent inspection by the NRC indicates that this review and evaluation was not done to the depth and detail necessary to resolve this matter.

In addition to those specific items discussed in the various functional areas, other items that indicated a lack of sufficient administrative controls and management attention during this assessment period include:

- The inadequate review of OPPD's response to IE Bulletin 82-02 and the subsequent escalated enforcement, as discussed in Section IV.C of this report, indicated a weakness in the licensee's technical evaluation and review process. Although the specific incident occurred before this SALP evaluation period, the investigation during this SALP period revealed that weaknesses still existed. The licensee exerted significant effort during the past nine months to correct these problems.
- The failure to followup on identified deficiencies in a QA audit of the plant security program, and the lengthy times involved in closing out other deficiency reports/quality reports indicated that management failed to achieve appropriate and timely corrective actions.

2. Conclusions

The licensee has demonstrated improvement in the QA area with regard to program implementation. OPPD's administrative controls and management attention had weaknesses in the specific areas outlined above and in the other analyses sections. The establishment of a separate Nuclear Production Division is a positive step towards bringing management controls to bear on those areas that need attention.

The licensee is considered to be in Performance Category 2 in this area.

Trend: Improved

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should be maintained at normal levels.

b. Recommended Licensee Actions

The licensee management needs to work toward a timely resolution of the NRC concerns regarding construction records and to address the other weaknesses identified.

J. Licensing Activities

1. Analysis

The NRC Office of Nuclear Reactor Regulation has performed an assessment of licensee performance in the functional area of licensing activities. Refer to Attachment 1 for details of this assessment.

2. Conclusions

As discussed in Attachment 1, the licensee is considered to be in Performance Category 1 in this area.

Trend: Improved

3. Board Recommendations

a. Recommended NRC Actions

Continue to perform licensing activities as required.

b. Recommended Licensee Actions

The licensee should continue its high level of management involvement in this area.

K. Training

1. Analysis

This functional area was inspected by region-based NRC inspectors and on a periodic basis by the NRC resident inspector. Three violations and no deviations were identified in this functional area during the appraisal period:

- Failure to follow approved training and qualification plan as it pertains to firearms qualification program. (Severity Level IV, 8326-02)
- Failure to complete security training program in accordance with commitment made to NRC. (Severity Level IV, 8326-03)
- Failure to provide training as required by the Fort Calhoun Station Security Plan. (Severity Level IV, 8417-01)

The licensee has devoted much attention to this functional area since receiving a Category 3 rating in the previous SALP report. An outside consultant was brought in for evaluation and consultation during the winter of 1983/84, and in May 1984, a working group was in place preparing training materials. This group started with about five persons and has grown to 15 during the past nine months. Following the failure of all three license candidates in June 1984, OPPD initiated an evaluation to identify and correct the specific causes of these failures. Part of this evaluation included an independent task force assessment of the existing operator training program. The results of this assessment formed the bases for short term and long term corrective actions to be implemented to ensure that future license candidates are prepared to pass the NRC examination and safely operate the plant. OPPD and NRC personnel met in September 1984, to discuss the licensee's operator licensing program and the results of this independent task force assessment. OPPD established the position of Supervisor-Training Services in July 1984, to head the newly established offsite Training Services Department. The NRC resident inspector reviewed the independent assessment report and concluded that the observations made represented an accurate picture of the training program at that time, and that the three groups of recommendations set forth would correct the problems if implemented. Some results from the efforts described above that occurred during this evaluation period included: (1) improved and expanded lesson plans and training packages, (2) the successful licensing of two reactor operator license candidates, (3) the addition of instructors from an inter-plant transfer and outside contractors, and (4) the appointment of the new Supervisor-Station Training. The NRC resident inspector reviewed portions of the revised student handout material and instructor lesson plans and found them to be comprehensive, well organized, and clearly written.

Three operator licensing examinations were administered during this evaluation period as tabulated below:

	SRO Candidates			RO Candidates		
	Total	Pass	Fail	Total	Pass	Fail
November 29-December 1, 1983	6	3	3	1	1	0
June 5-7, 1984	0	-	-	3	0	3
November 7-8, 1984	0	-	-	2	2	0

One requalification examination was administered as tabulated below:

	SRO Candidates			RO Candidates		
	Total	Pass	Fail	Total	Pass	Fail
November 7, 1984	5	2	3	2	2	0

The first two examination results continued a downtrend that was identified in the previous SALP period. These results were below the industry norm and reflected a weakness in the licensee's training department to adequately screen and prepare candidates for licenses. The intensive training effort focused on the November 1984, candidates and the successful licensing of both examinees indicated that this trend may have been reversed.

License requalification training continued to be a problem area during this evaluation period. Interviews by the NRC resident inspector and a region based NRC inspector with licensed operators indicated that actual in-class lecture time was being cut short and that the quality of training was suffering because of the limited availability of experienced licensed instructors. In addition, the high failure rate for new license candidates and NRC requalification candidates during this appraisal period seemed to confirm that the overall quality of training at the licensed operator level remained marginal. The NRC resident inspector attended selected training lectures during this appraisal period and observed a wide variation in quality of handout material and instructor capabilities.

As part of the overall restructuring of the training program, a new "performance based" training program for newly hired auxiliary operator trainees has been developed and will be implemented during the first quarter of 1985. This program has clearly defined written goals, a complete schedule for classroom and onshift time, and a structured system of lesson plans, qualification cards, and practical factors to be completed. This program is one of many being developed by the licensee to qualify for INPO certification in 1986.

The in-house training program for five inexperienced chemistry/health physics technicians was completed this evaluation period and all five trainees became qualified to work shift-technician duties. It was noted in an NRC inspection report that the training department did not have an instructor qualified or experienced in nuclear power plant chemistry and that a comprehensive training program for radwaste operators did not exist. The chemistry/radiochemistry training program was being conducted primarily by the chemistry section senior staff personnel under the supervision of the plant chemist. A review of qualification verification records for Chemistry and Radiation Protection supervisory personnel revealed that no training had been completed in any of the study areas listed in Section 6.1 of the Fort Calhoun Station Training Manual.

The NRC resident inspector noted that the training manual prepared for Cycle 9 modifications was the most complete and informative of those issued to date. The licensee provided acceptable training on short notice in response to the steam generator tube failure incident and covered the applicable emergency procedures, the lessons learned from the Ginna tube rupture incident, and the revised Technical Specification/Surveillance Tests.

2. Conclusions

The licensee has done an excellent job of identifying and evaluating the problems in this functional area, and has expressed a strong commitment to resolve these matters. The initial results of their efforts have been positive, but the overall training program has not yet seen the benefits of OPPD's commitments and plans. During this evaluation period the licensee's record in licensee examinations and requalification examinations remained poor. Many of the personnel changes to the training department and the appointment of the Supervisor-Station Training occurred too late in this SALP evaluation period to have had a significant impact on performance.

The licensee is considered to be in Performance Category 3 in this functional area.

Trend: Improved

3. Board Recommendations

a. Recommended NRC Actions

The NRC inspection effort in this functional area should be increased. An evaluation of the licensee's corrective actions and their impact on the training program should be performed by August 31, 1985.

b. Recommended Licensee Actions

Licensee management should provide aggressive action to ensure control of the training program in order to maintain the positive trend that appears to have been established toward the end of this evaluation period. The effort at upgrading lesson plans and student material needs to be carried through to completion. The need for an expanded training staff remains and this shortcoming should be resolved as soon as possible. The licensee should further consider the benefits of obtaining access to a site-specific simulator for training of the Fort Calhoun Station operators.

The quality of instructors should be evaluated and training provided as required to increase the effectiveness of the training staff. Training in the areas of security, chemistry, and radiation protection needs to be better coordinated and administered under the training department.

V. SUPPORTING DATA AND SUMMARIES

A. Licensee Activities

1. Major Outages

The refueling outage occurred during the period March 5 to July 8, 1984. In addition to the insertion of 40 new fuel bundles into the core, major planned maintenance activities included eddy current testing of both steam generators, sludge lancing of the secondary side of both steam generators, repair of the reactor coolant pump gaskets, work on the QSPDS, and replacement of the high pressure turbine rotor. Three additional activities that occurred during this outage included the "rim cut" on the steam generator's tube support plate, plugging of the failed tube in Steam Generator RC-2B, and rework of 638 containment penetration lead wires to protect them from the harsh environment of a LBLOCA.

Fort Calhoun Station experienced a 2-week outage from November 18 to December 3, 1984, to repair a body-to-bonnet flange leak on a pressurizer spray valve.

2. Power Limitations

The reactor was not limited in power level below the licensed limits during this appraisal period.

3. License Amendments

Amendment No. 75	Authorized Spent Fuel Pool Rerack, September 9, 1983
Amendment No. 76	Administrative Changes, January 26, 1984

Amendment No. 77	Cycle 9 Restart, April 26, 1984
Amendment No. 78	Shift Manning and QC Personnel Changes, May 16, 1984
Amendment No. 79	Snubber Changes, May 23, 1984
Amendment No. 80	Add Operability and Surveillance Requirements for RCS Vents and Administrative Requirements for Analysis of Plant Effluents, July 9, 1984
Amendment No. 81	Add Operability and Surveillance Requirements for Containment Wide Range Radiation Monitors, Wide Range Noble Gas Monitors, and Main Steam Lines Radiation Monitor, July 12, 1984
Amendment No. 82	Add Operability and Surveillance Requirements for Containment Hydrogen, Water, and Pressure Monitors, August 2, 1984
Amendment No. 83	Update Surveillance Capsules Removal Schedule, September 7, 1984
Amendment No. 84	Plant Support and Plant Organization Changes, September 7, 1984
Amendment No. 85	Limit Overtime and Report PORV/SV Failures and Challenges, October 11, 1984

4. Significant Modifications

Major modifications completed during this appraisal period included the installation of new spent fuel racks, removal of steam generators drilled tube support plate rim, implementation of new secondary chemistry control in response to the failed tube in Steam Generator RC-2B, and the upgrading of instrumentation, limit switches, containment penetrations, etc. to meet EEQ requirements.

B. Inspection Activities

1. Violations

See Table 1.

2. Major Inspections

During this appraisal period, one special inspection was conducted in the area of security and safeguards. The inspection was performed by a team of three inspectors and one observer from outside Region IV and involved a total of 235 direct inspection man-hours. (NRC Inspection Report 50-285/84-20)

C. Investigations and Allegations Review

One investigation was conducted during this appraisal period and it addressed the material false statement made by the licensee in their response to IE Bulletin 82-02, "Degradation of Threaded Fasteners in Reactor Coolant Pressure Boundaries of PWR Plants." It was confirmed by the investigation that the OPPD response to the NRC was false in that "Super-Moly" (molybdenum disulfide) was used on the reactor vessel and reactor coolant pump studs and that a mixture of 50 percent oil and 50 percent graphite was designated for use on manway studs. These failures by the licensee to perform an adequate review of the related documentation, to coordinate the response with knowledgeable personnel, and to identify the false statement during the OPPD required procedural review, resulted in a Severity Level III Violation. An enforcement conference was held on December 20, 1984, between the NRC and the licensee to discuss this matter.

One allegation was received during this appraisal period that identified certain incidents which had occurred over the past five years and alleged poor management practices in the area of supervision and discipline that could affect the safety and health of the public. The review and followup of this allegation was still in progress at the close of this appraisal period, and is expected to be resolved during the first quarter of the next SALP period.

D. Escalated Enforcement Actions

1. Civil Penalties

Two notices of violation with proposed imposition of civil penalties were issued to the licensee during this appraisal period.

- A Severity Level III Violation and a proposed civil penalty of $40,000 were issued as a result of the material false statement made by the licensee in their response to IE Bulletin 82-02, "Degradation of Threaded Fasteners in Reactor Coolant Pressure Boundaries of PWR Plants." In consideration of OPPD's prior good performance in this area, and their prompt and extensive corrective actions, the Regional Administrator determined that the civil penalty should be fully mitigated.

- A Severity Level III Violation was issued in the area of security and safeguards that comprised a composite of seven Severity Level IV and V Violations and reflected an overall weakness in the Fort Calhoun Station security program. A civil penalty of $50,000 was proposed, but this amount was mitigated to $25,000 on the basis of the licensee's previous good enforcement history in this area.

E. Management Conferences Held During Appraisal Period

1. Conferences

The following conferences were held between Region IV and the licensee during this appraisal period:

- Enforcement conference of September 9, 1983, at the Region IV office to discuss NRC concerns related to construction QA records and the QA records file room. The bases for this meeting were the findings described in NRC Inspection Report 50-285/83-17.
- Management meeting of September 21, 1984, at the Region IV office to discuss the licensee's operator licensing program in response to Mr. J. T. Collins letter of August 13, 1984, to Mr. W. E. Miller of OPPD.
- Enforcement conference of October 11, 1984, at the Region IV office to discuss the results of NRC Inspection Report 50-285/84-20 which documented nine violations and two deviations identified by the special inspection team.
- Enforcement conference of December 20, 1984, at the Region IV office to discuss the licensee's response to IE Bulletin 82-02 and the associated material false statement cited in NRC Inspection Report 50-285/84-12.

2. Confirmation of Action Letters (CALs)

The following CALs were issued by Region IV during this appraisal period:

- J. T. Collins letter of June 5, 1984, to W. C. Jones of OPPD to confirm the actions and conditions required of OPPD in relation to the failed tube in Steam Generator RC-2B.
- J. T. Collins letter of August 16, 1984, to Mr. R. L. Andrews of OPPD to confirm the actions required of OPPD in response to security matters identified in NRC Inspection Report 50-285/84-17.

F. Review of Licensee Event Reports and 10 CFR Part 21 Reports Submitted by the Licensee

1. Licensee Event Reports (LERs)

The SALP Board reviewed the LERs for the period September 1, 1983, through February 28, 1985. This review included LERs 83-008 through 83-013, and 84-001 through 84-025. The SALP Board reviewed the licensee's cause classification for these LERs and did not identify any significant differences between those made by the licensee and those made independently by the board.

Due to the revised LER rule that went into effect on January 1, 1984, the licensee was required to report a significant number of unplanned VIAS actuations that were not performing a safety function. This resulted in an increase of LERs this appraisal period even though the new rule was intended to eliminate inconsequential reports. The licensee is considering a Technical Specification revision to modify this specific LER requirement.

The NRC Office for Analysis and Evaluation of Operational Data performed a review of licensee LERs, focusing on the accuracy and completeness of the reports. Refer to Attachment 2 for details of this review.

2. Part 21 Reports

None

G. NRR Activities

1. NRR License Meetings

December 20, 1983	SALP
March 23, 1984	Environmental Qualification
April 17, 1984, and October 11 & 13, 1983	Radiological Effluent Technical Specifications
May 29, 1984	Steam Generator B Major Leakage Event
December 13, 1984	Plant Security
February 5-8, 1985	In-Progress Audit of Licensee's Detailed Control Room Design Review

2. NRR Site Visits

October 11 & 14, 1983	Discussed Licensing Actions with Resident Inspector and Visited Local PDR
May 23-26, 1984	Emergency Trip to Address Steam Generator B Major Leakage Event
August 27-29, 1984	Toured Plant, Reviewed TMI Related Modifications, and Discussed Licensing Actions with Resident Inspector
February 6-7, 1985	Toured Control Room and Remote Shutdown Panel and Discussed Licensing Actions with Resident Inspector

3. Commission Briefings

None

4. Schedular Extensions Granted

IST 2nd 10-year program, interim schedular relief for 1 year, September 30, 1983

IST 2nd 10-year program, interim schedular relief for 1 year, October 9, 1984

EQ Schedular Extension, May 18, 1984

5. Reliefs Granted

ISI 1st 10-year program, 2 reliefs, November 14, 1984

ISI 2nd 10-year program, 1 relief, September 30, 1983

ISI 2nd 10-year program, 8 reliefs, April 6, 1984

6. Exemptions Granted

None

7. Emergency Technical Specifications Issued

None

8. Orders Issued

   Order confirming licensee commitments on emergency response capability as required by Supplement 1 to NUREG-0737, February 22, 1984.

9. NRR/Licensee Management Conferences

   None

TABLE 1

INSPECTION ACTIVITY AND ENFORCEMENT

FUNCTIONAL AREA	NO OF VIOLATIONS IN EACH SEVERITY LEVEL					
	V	IV	III	II	I	DEVIATIONS
A. Plant Operations		4				
B. Radiological Controls	1					
C. Maintenance	2	2				
D. Surveillance						
E. Fire Protection						
F. Emergency Preparedness						
G. Security and Safeguards		5	1*			3
H. Refueling						
I. Quality Programs and Administrative Controls Affecting Quality	1	4	1			
J. Licensing Activities						
K. Training		3				
TOTAL	4	18	2*	0	0	3

*This comprises seven Severity Level IV and V Violations identified in NRC Inspection Report 285/84-20.

Attachment 1

Docket No. 50-285

FACILITY: Fort Calhoun Station, Unit No. 1

LICENSEE: Omaha Public Power District

EVALUATION PERIOD: September 1, 1983 to February 28, 1985

PROJECT MANAGER: Ed Tourigny

I. INTRODUCTION:

This report contains NRR's input to the SALP review for the Fort Calhoun Station, Unit No. 1. The assessment of the licensee's performance was conducted according to NRR Office Letter No. 44, NRR Inputs to SALP Process, dated January 3, 1984. This Office Letter incorporates NRC Manual Chapter 0516, Systematic Assessment of Licensee Performance.

II. SUMMARY

NRC Manual Chapter 0516 specifies that each functional area evaluated will be assigned a performance category (Category 1, 2 or 3) based on a composite of a number of attributes. The performance of the Omaha Public Power District in the functional area of Licensing Activities is rated Category 1

III. CRITERIA

The evaluation criteria used in this assessment are given in NRC Manual Chapter 0516 Appendix, Table 1, Evaluation Criteria with Attributes for Assessment of Licensee Performance.

IV. METHODOLOGY

This evaluation represents the integrated inputs of the Operating Reactor Project Manager (ORPM) and those technical reviewers who expended significant amounts of effort on the Fort Calhoun Station licensing actions during the current rating period. Using the guidelines of NRC Manual Chapter 0516, the ORPM and each reviewer applied specific evaluation criteria to the relevant licensee performance attributes, as delineated in Chapter 0516, and assigned an overall rating category (1, 2 or 3) to each attribute. The reviewers included this information as part of Safety Evaluation Reports transmitted to the Division of Licensing. The ORPM, after reviewing the inputs of the technical reviewers, combined this information with his own assessment of licensee performance and, using appropriate weighting factors, arrived at a composite rating for the licensee. This rating also reflected the comments of the NRR Senior Executive assigned to the Fort Calhoun Station, Unit No. 1 SALP assessment. A written evaluation was then prepared by the ORPM and circulated to NRR management for comments which were incorporated in the final draft.

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The basis for this appraisal was the licensee's performance in support of licensing actions that were either completed or had a significant level of activity during the current rating period. These actions, consisting of amendment requests, exemption requests, relief requests, responses to generic letters, TMI items, and other actions, are classified as follows:

- 10 Multi-Plant Actions (9 completed). Included in this category are

- EQ of Safety Related Electrical Equipment, B-60, Completed
- Masonry Wall Design Review, B-59, Completed
- Adequacy of Station Electrical Distribution Voltages, B-48, Completed
- Control of Heavy Loads, Phase I, C-10, Completed
- Asymmetric LOCA Loads, D-10, Completed
- Plans for Preventing Exceeding PTS Screening Criteria, B-73, Completed
- Natural Circulation Cooldown, B-66, Completed
- TSs In Response to GL-82-16, NUREG-0737, B-72, Completed
- TSs In Response to GL-82-37, NUREG-0737, B-83, Completed
- Appendix I Tech.Spec.Implementation Review, A-02

- 24 Plant Specific Actions (17 Completed). Included in this category are

- Evaluation of Neutron Source Data for Neutron Flux Reduction Verification, Completed
- Relief from ISI 1st Ten Year Program, Completed
- Second Interim Schedular Relief From IST 2nd Ten Year Program, Completed
- Plant Support and Plant Organizational TS Changes, Completed
- Update Surveillance Capsules Removal Schedule TS and Update Snubber Listing TS, Completed
- Steam Generator B Major Leakage Event Evaluation, Completed
- Evaluation of Licensee's Position on AFW System Pump Testing Frequency, Completed
- Evaluation of Methodology Reports for Reloads, Completed
- First EQ Deadline Time Extension, Completed
- Evaluation of SCU Methodology for Reloads, Completed
- Shift Manning and QC Personnel TS Changes, Completed
- Thermal Shield Review, Completed
- Cycle 9 Startup Evaluation, Completed
- Evaluation Related to GL-82-17 and GL-82-23, Completed
- Administrative TS Changes, Completed
- Spent Fuel Pool Rerack TS Changes, Completed
- First Interim Schedular Relief From IST 2nd Ten Year Program, Completed
- TS on Inoperability of RPS/ESFAS Channels
- Appendix R Exemption Requests (Outside Containment)
- Appendix R Exemption Requests (Inside Containment)
- Steam Generator Mid-Cycle Tube Inspection Waiver
- Re-Evaluation of Licensee's Position on AFW System Pump Testing Frequency
- Second EQ Deadline Time Extension
- Administrative TS Changes as a Result of Rule Changes

- 14 TMI (NUREG-0737) Actions (8 Completed). Included in this category are:

- Thermal Mechanical Report, F-30, Completed
- Auto PORV Isolation, F-36, Completed
- Report on PORV Failures, F-37, Completed
- Potential For Voiding in RCS, F-33, Completed
- RCS High Point Vents, F-10, Completed
- Post Accident Sampling Modifications, F-12, Completed
- NUREG-0737 Supplement 1 Confirmatory Order, Completed
- Detailed Control Room Design Review Program Plan, F-08, Completed
- Inadequate Core Cooling Instrumentation, F-26
- Detailed Control Room Design Review Summary Report Including In-Progress Audit, F-71
- Safety Parameter Display System, F-9
- Technical Support Center, F-63
- Operation Support Center, F-64
- Emergency Operations Facility, F-65

V. ASSESSMENT OF PERFORMANCE ATTRIBUTES

The licensee's performance evaluation is based on consideration of seven attributes as specified in NRC Manual Chapter 0516. For most of the licensing actions considered in this evaluation, only three of the attributes were of significance. Therefore, the composite rating is heavily based on the following attributes:

- Management Involvement and Control in Assuring Quality
- Approach to Resolution of Technical Issues from a Safety Standpoint
- Responsiveness to NRC Initiatives.

With the exception of Enforcement History, for which there was no basis within NRR for evaluation, the remaining attributes of

- Reporting and Analysis of Reportable Events
- Staffing (Including Management)
- Training and Qualification Effectiveness

were judged to apply only to a few licensing activities.

A. Management Involvement and Control in Assuring Quality

The licensee's management has demonstrated a high level of involvement and control in assuring quality. Regarding multi-plant actions, the licensee's management became highly involved in resolving issues such as Environmental Qualification of Electrical Equipment, Adequacy of Station Electrical Distribution Voltages, Control of Heavy Loads - Phase 1, and Pressurized Thermal Shock Screening Criteria. Although the technical specifications for Appendix I have not been implemented yet, the licensee's management was heavily involved in resolving the issues. There is currently one multi-plant action for which we believe a higher level of management involvement and control is necessary. It deals with masonry walls. Plant modifications regarding masonry walls should have been done by December 31, 1984. The current schedule calls for all work to be done by the end of the 1985 refueling outage.

In regard to management involvement and control insofar as plant specific actions are concerned, the licensee's management took appropriate steps in resolving most of the issues. The licensee's management assured that the station had a sound inservice inspection program for the second ten year period. In those cases where the licensee determined the ASME Code to be impractical, relief was requested. The licensee's management realized that they had a problem in environmentally qualifying certain electrical equipment, and asked for a time extension. When the licensee's management realized that the equipment could not be qualified by testing, they took immediate steps to modify the equipment that would be needed for the large break loss of coolant accident and requested a time extension for the remaining equipment. Technical specification changes in relationship to reloads have become more complicated over the years. In order to permit the reviews associated with reloads to become more straightforward, the licensee submitted methodology reports on reloads. These were reviewed and approved by the staff. As a result, the reload review performed during the evaluation period was more straightforward. Future reload reviews should also be more straightforward. A major steam generator tube leakage occurred during the evaluation period, and the licensee's management was highly involved in resolving all the issues. The licensee's management took an aggressive approach in addressing the generic thermal shield problem. Although no problems were found with the Fort Calhoun thermal shield, the licensee's management committed to perform a detailed thermal shield inspection no later than the 1987 refueling outage.

There were a number of plant specific actions where the licensee's management took significant steps in resolving, although the reviews are not yet complete. These dealt with the inoperability of RPS/ESFAS channels and the Appendix R exemptions. These reviews should be completed in the next evaluation period. There are a number of other active plant specific actions where the licensee's management could have become more involved, e.g. the testing frequency of the auxiliary feedwater pumps, upgrading the plant's technical specifications to be consistent with the new LER rule. We expect resolution on these issues during the next evaluation period.

In regard to management involvement and control as far as TMI actions are concerned, the licensee's management has taken many steps in getting the systems installed and operational. Examples include RCS vents, PASS, containment monitors for hydrogen, water, radiation and pressure, and toxic gas monitors. In addition, many of the NUREG-0737 technical specification changes have been applied for by the licensee and have been issued by the staff. Examples include RCS vents and containment monitors for hydrogen, water, radiation and pressure. We hope to see the remaining NUREG-0737 systems installed and made operational in a timely manner and the technical specification change applications applied for in a reasonable period of time after declaring the systems operational. Many of the systems are under confirmatory order to be made operational by agreed to dates.

Based upon the above evaluation, a rating of category 1 is assigned to this attribute.

B. Approach to Resolution of Technical Issues from a Safety Standpoint

The licensee has demonstrated resolution of technical issues from a safety standpoint. Regarding multi-plant actions, many difficult licensing issues were resolved during the evaluation period. Examples include Environmental Qualification, Control of Heavy Loads - Phase 1, and Pressurized Thermal Shock Screening Criteria. Although the technical specifications for Appendix I have not been implemented yet, the technical issues have been resolved.

In regard to resolution of technical issues insofar as plant specific items are concerned, many major licensing issues were resolved during the evaluation period, and many are near resolution. The major ones resolved include Steam Generator Integrity, Reload Methodology, Thermal Shield Integrity, and Spent Fuel Pool Reracking. A major steam generator tube leakage occurred during the evaluation period. Much staff/licensee technical interface occurred before the licensee was authorized to restart the Station. The crux of the issue was what caused the leakage and would it happen again. It was determined that intergranular stress corrosion cracking was the cause of the failure. It is impossible to state whether it will happen again; however, the licensee inspected almost all the tubes in both generators, performed a double verification of the results, upgraded the secondary water chemistry program, implemented an enhanced condenser inspection and repair program, and committed to a more detailed steam generator inspection program during the next refueling outage. The number of tubes to inspect was a technical issue in the earlier stages of the review and the licensee agreed to inspect all that were accessible by the remote probe insertion machine. As stated in the previous evaluation on management involvement, reload technical specification changes became more difficult to review. Much technical work was performed on the part of the licensee to develop and implement reload methodology reports. Much technical work was performed by the staff to review and approve the reports. We believe that the processing of reload applications will be more straightforward in the future because the licensee now has an approved methodology and the staff has a good appreciation on how the licensee performs the safety analyses. The integrity of the licensee's thermal shield was a tough licensing issue during the evaluation period. Although the shield was demonstrated by inspection to be in good condition and the licensee provided analysis to show that if the shield failed during operation no safety question would arise, the licensee committed to perform another detailed inspection no later than the 1987 refueling outage to prove that the shield continues to be in good condition. The authorization to rerack the spent fuel pool was issued early in the evaluation period and the storage capacity assuming full core discharge capability exists until the year 1996. It should be noted that this was one of the first authorizations to rerack the spent fuel pool using two fuel zones: one fuel zone containing neutron poison material (boraflex) and one fuel zone without neutron poison material.

The plant specific issues that are nearly complete are the technical specifications on RPS/ESFAS channel inoperability and Appendix R exemption requests. Technical issues still remain on the auxiliary feedwater system pump testing frequency; it is hoped that this technical issue will be resolved during the next evaluation period. It is believed that the licensee can develop a better rationale for testing the pumps on a quarterly frequency versus the staff requested monthly frequency.

In regard to resolution of technical issues relating to TMI actions, the licensee has resolved all the issues discussed to date and, as stated previously, many of the technical specifications are in place. The bulk of the outstanding TMI related work should be completed in the next evaluation period. It was stated previously that the license is under a confirmatory order to complete most of these items.

On the basis of the above observations, a rating of category 1 is assigned to this attribute.

C. Responsiveness to NRC Initiatives

The licensee is responsive to NRC initiatives. The licensee realized early in the evaluation period that many licensing actions were outstanding for a number of years and committed to cleaning up the backlog. We are pleased to say that much of the backlog was cleaned up. Regarding multi-plant actions, all of the backlog which the licensee had control over was cleaned up except the Appendix I technical specifications. It should be noted that the technical and managerial issues are resolved for Appendix I; all that needs to be done is for the staff to issue the technical specifications. It should also be noted that there are some backlog multi-plant actions over which the licensee had no control during the evaluation period. These include thirteen reviews associated with the Salem ATWS event (GL-83-28). The licensee submitted the responses during the evaluation period on the requested schedule; the responses are under review.

Insofar as plant specific actions are concerned, many of these were completed during the evaluation period and some are near completion. The only criticism we have regarding the licensee's responses to NRC initiatives are the issues on the auxiliary feedwater (AFW) system pump testing frequency and the technical specifications relating to the LER rule. A letter was sent to the licensee in October 1984 requesting a meeting to discuss and resolve the AFW system pump testing frequency issue. The licensee has not responded in writing as of February 28, 1985, the end of the evaluation period. In addition, the licensee was requested in May 1984 to submit more straightforward technical specifications regarding the LER rule. The licensee has not responded as of February 28, 1985, the end of the evaluation period.

Regarding TMI items, the licensee has placed many of the systems in operation and the technical specifications for many of the systems were put in place during the evaluation period. The bulk of the remaining systems are under confirmatory order to be made operational during the next evaluation period. Most of the licensee's responses to NRC initiatives in the TMI area have been completed, and the remaining completions are in sight.

In order to provide higher quality submittals to the staff, the licensee took the initiative and established a separate review group during the evaluation period. This group reviews all significant licensee submittals for accuracy. However, we have noted in some instances that the licensee submittals have been delayed because of this augmented process. We expect such cases to be the exception and not the rule.

On the basis of the above considerations, a rating of category 2 is assigned to this attribute.

D. Enforcement History

No basis exists for an NRR evaluation of this attribute.

E. Reporting and Analysis of Reportable Events

The licensee continues to keep the NRR staff informed in a timely manner of all significant events at the plant whether they are reportable or not. Although reportable events are discussed in detail elsewhere in the NRC evaluation, we believe that NRR comments are worthwhile, especially when they highlight good licensee practices. Examples of events that were reported to the staff were a control rod drive mechanism operability problem, minor primary to secondary leakage in steam generator B, major failure of a tube in steam generator B, transformer fire in the switchgear room, temporary loss of the 161 kV line, gaseous radioactivity release in the auxiliary building, spent fuel pool reracking problem, and various VIAS actuations. Some of these events were not reportable but demonstrate that the licensee keeps the staff informed of significant events. The staff appreciates such information and hopes that this practice will continue.

On the basis of the above observations, a rating of category 1 is assigned to this attribute.

F. Staffing (Including Management)

The licensee has shown significant initiative in the staffing area during the evaluation period. Although staffing is discussed in other sections of the NRC evaluation, we believe that NRR comments are worthwhile since, in this case, the initiatives are highly noteworthy.

The licensee implemented a higher level of corporate oversight of the Fort Calhoun Station during this evaluation period. In the past, the plant managers for the coal and nuclear electrical production stations reported through the Section Manager of Operations to the Division Manager, Production Operations. A new Division Manager of Nuclear Production position was authorized and filled. The new Division Manager is responsible for the Fort Calhoun Station, and the Fort Calhoun Station plant manager reports directly to him. We have noted that since the change was made, the Division Manager has become more involved in the management of the plant and issues related to the plant. As an example, we have noted that the Division Manager visits the plant on a higher frequency and interacts more with the plant staff. As another example, we have noted more involvement of the Division Manager in significant licensing issues. The Station's technical specifications were amended during the evaluation period to reflect this new initiative.

The licensee implemented new staffing initiatives at the Fort Calhoun Station also. Two new positions have been added to the staff of the Supervisor-Chemical and Radiation Protection: Radioactive Waste Coordinator and ALARA Coordinator. These positions have been filled. The Supervisor-Training position was elevated to the position of Supervisor-Station Training and this person now reports to the station manager. More full time trainers were also added to the Station complement. The licensee took steps to ensure that it met the upgraded NRC regulations on shift manning that became effective on January 1, 1984. The licensee did not have to request an exemption. Lastly, the Quality Control (QC) personnel reported to the Maintenance Supervisor. They now report to the

Technical Supervisor. This change allowed the QC personnel to have a higher level of independence from the maintenance department. The plant's technical specifications were amended during the evaluation period to reflect these new initiatives.

The licensee also implemented new staffing initiatives in their corporate headquarters to further support the Fort Calhoun Station staff. As an example, additional resources were devoted to handle licensing issues.

Based on the above considerations, a rating of category 1 is assigned to this attribute.

VI. CONCLUSION

A complete performance rating of category 1 has been assigned by the NRR SALP evaluation effort for the rating period.

Enclosure 2

Information to be added to
Section V of SALP Report -
"Supporting Data and Summary"

1. NRR/licensee Meetings

February 5-8, 1985,	In-Progress Audit of Licensee's Detailed Control Room Design Review
December 13, 1984,	Plant Security
May 29, 1984,	Steam Generator B Major Leakage Event
April 17, 1984 and October 12-13, 1983,	Radiological Effluent Technical Specifications
March 23, 1984,	Environmental Qualification
December 20, 1983,	SALP

2. NRR Site Visits

February 6-7, 1985,	Toured Control Room and Remote Shutdown Panel and Discussed Licensing Actions with Resident Inspector
August 27-29, 1984,	Toured Plant, Reviewed TMI Related Modifications, and Discussed Licensing Actions with Resident Inspector
May 23-26, 1984,	Emergency Trip to Address Steam Generator B Major Leakage Event
October 11 and 14, 1983,	Discussed Licensing Actions with Resident Inspector and Visited Local PDR.

3. Commission Briefings

None.

4. Schedular Extensions Granted

IST 2nd 10 year program, interim schedular relief for one year, October 9, 1984

IST 2nd 10 year program, interim schedular relief for one year, September 30, 198

EQ Schedular Extension, May 18, 1984

5. Reliefs Granted

ISI 1st 10 year program, 2 reliefs, November 14, 1984

ISI 2nd 10 year program, 8 reliefs, April 6, 1984

IST 2nd 10 year program, 1 relief, September 30, 1983

6. Exemptions Granted

None.

7. License Amendments Issued

Amendment No. 85	Limit Overtime and Report PORV/SV Failures and Challenges, October 11, 1984
Amendment No. 84	Plant Support and Plant Organization Changes, September 7, 1984
Amendment No. 83	Update Surveillance Capsules Removal Schedule, September 7, 1984
Amendment No. 82	Add Operability and Surveillance Requirements for Containment Hydrogen, Water, and Pressure monitors, August 2, 1984
Amendment No. 81	Add Operability and Surveillance Requirements for Containment Wide Range Radiation Monitors, Wide Range Noble Gas Monitors, and Main Steam Lines Radiation Monitor, July 12, 1984
Amendment No. 80	Add Operability and Surveillance Requirements for RCS Vents and Administrative Requirements for Analysis of Plant Effluents, July 9, 1984
Amendment No. 79	Snubber Changes, May 23, 1984
Amendment No. 78	Shift Manning and QC Personnel Changes, May 16, 1984
Amendment No. 77	Cycle 9 Restart, April 26, 1984
Amendment No. 76	Administrative Changes, January 26, 1984
Amendment No. 75	Authorized Spent Fuel Pool Rerack, September 9, 1983

8. Emergency Technical Specifications Issued

None.

9. Orders Issued

Order confirming licensee commitments on emergency response capability as required by Supplement 1 to NUREG-0737, February 22, 1984

10. NRR/Licensee Management Conferences

None.

Attachment 2

AEOD INPUT TO SALP REVIEW FOR FORT CALHOUN

The licensee submitted about 30 reports, plus updates, during the assessment period from September 1, 1983 to February 28, 1985. Our review included the following LER numbers:

83-008 to 83-013
84-001 to 84-025

The LER review followed the general instructions and procedures of NUREG-0161 and NUREG-1022. The specific review criteria and our findings follow:

1. LER Completeness

   a) Was the information in the LER Form sufficient to provide a good understanding of the event?

   1983 LERs

   The information in the two free-form narrative sections of the LER Form provided sufficient information for a clear and useful description of the occurrence, the direct consequences and the corrective action. The abstracts typically included specific details of the event such as valve identification numbers, model numbers, number of operable redundant systems, the date of completion of repairs, etc., to provide a good understanding of the event. The reports were easy to read and meaningful.

   Several LERs had overrunning narratives that might be a problem for future abstracting. Overrunning narratives are so unnecessary for Fort Calhoun because the supplemental information in the attachments to the LER Form are so consistently well written.

   1984 LERs

   The abstract described the major occurrences of the event, including all component or system failures that contributed to the event and the significant corrective actions taken or planned to prevent recurrence as stated in NUREG-1022.

b) Were the LERs coded correctly?

1983 LERS

We checked the codes that the licensee selected against the narrative description of the event for accuracy. We agreed with the licensee's entry in all coded fields.

1984 LERs

The above comments are applicable.

c) Was supplemental information provided when needed?

1983 LERs

Every report contained additional supplementary information in the form of attachments. The attachments were titled and consistently arranged with Attachment No. 1 as the safety analysis, Attachment No. 2 as the the corrective action, and Attachment No. 3 as the failure data.

The separate titles with specific information in each section led to a consistency from LER to LER that is not approached by other schemes of providing supplemental information. The safety analysis, even for minor events, was relevent and comprehensive. Perhaps it is simply easiest to say that they are the best written and most informative LERs that I have reviewed. These reports best represent how LERs should be written.

1984 LERs

The above comments are also applicable for 1984 LERs. The supplemental information is thorough, completely detailed and well written. They simply provide all the information that is required, LER after LER. We have no lingering questions about the event after reviewing one of these LERs.

d) Follow-up Reports

1983 LERs

The licensee only updated one report: LER 83-008. The area of new narrative information was stated by the licensee in the transmittal letter, but the coding rules of NUREG-0161 for updating LERs were not completely followed.

1984 LERs

Only LER 84-008 was updated. It provided new information and the portion that was revised was denoted by a vertical line in the right hand margin so the extent of the revision could easily be determined by the reader.

e) Were similar occurrences properly referenced?

1983 and 1984 LERs

The licensee stated the judgement criteria used to define similar events, the number of times the similar event occurred, and the previous LER numbers. In addition, when there have been no previous events, the licensee positively states that this is the first reportable occurrence of this type.

2. Multiple Event Reporting in a Single LER

The licensee submitted many LERs that combined multiple events of component actuations into a single report. These multiple events were combined correctly into a single LER in accordance with the guidelines of NUREG-0161 and NUREG-1022.

3. Prompt Notification Follow-up Reports

Five PNs were issued in this SALP assessment period. Three of the PNs described clearly unreportable events. The remaining two PNs were covered by multiple event LERs on the ventilation isolation actuation system, so the licensee appears to be reporting all events that are reportable.

Mrs. **Smith**. I would like to hear your comments about the fact that Cuba is going to have a nuclear reactor on line before too long.

Do you think we will know if something happened down there?

Mr. **Dicks**. Admiral, have you visited that one?

Mr. **Zech**. No, sir, I haven't; probably won't.

Mr. **Palladino**. Mrs. Smith, I don't think we will have any say in that.

Mrs. **Smith**. Are you concerned about it?

Mr. **Palladino**. Yes. I am also concerned about reactors that are built in countries where I don't know what their standards are, but we have had very little technical exchange with the Russians since they went into Afghanistan.

Mrs. **Smith**. The others in this hemisphere, do you know anything about them?

Mr. **Palladino**. Generally, we know quite a bit about the reactor designs that the free world is working on.

There is free exchange of information. There is also exchange of design basis information, codes and the like.

Mr. **Denton**. I can add a little bit.

The focus of the task force has not been on such matters but we were questioned about it.

It is our understanding the plant in Cuba is not a graphite moderated pressure tube reactor. It is more a light water reactor such as they exported to Finland, for example.

In the course of getting into the export market, that type reactor is one I have visited; that does have a U.S.-type containment on it.

So, specifically I do not think the reactor being built in Cuba is comparable to the Chernobyl reactor. I think it is more like the U.S. plants.

Mrs. **Smith**. Thank you very much. I have no more questions, Mr. Chairman.

Mr. **Bevill**. The Chair recognizes Mr. Fazio.

Mr. **Fazio**. Thank you, Mr. Chairman.

I will be brief because I have to get over to the Budget Committee meeting where they are trying to decide how high the user fees have to be in order to provide for your budget. I would like to just get a general impression very quickly and I hate to be confrontational.

That is not my real purpose. But since the AEC breakup in the early seventies, there has been perhaps some reduction in the ability for you and for the DOE to know exactly what each other is doing. I would like your evaluation of the quality of the safety that is being employed at our defense reactors.

I would like your impression as to whether or not you feel if we were, for example, to build a new plutonium production facility that we should or should not have containment or whether we should go to confinement. I would be interested in your evaluation as to whether or not you think the operation of the existing facilities should be extended. Should we be moving toward newer and safer facilities.

I would be interested in hearing from those who want to comment briefly on this general area.

Mr. **Palladino**. Mr. Fazio, I will start.

My feeling, my impression is that DOE has very competent people. They know these designs very well. They have long experience.

As a matter of fact, much of the experience in the early days really were in what I will call the DOE-type facilities. So they are not neophytes in this business.

They are quite competent, quite experienced. They have different missions in life from the NRC. They have to provide weapons material primarily.

So they have to weigh a lot of factors that we don't have to weigh. Now, I am not saying that an independent review wouldn't always help.

It helps us when we get them.

Mr. FAZIO. Do you think it would help?

Mr. PALLADINO. In the early days they did have the advantage of ACRS comments, but those seemed to cease in 1974.

Mr. FAZIO. Do you think it would help at the present time to have an independent review?

Mr. PALLADINO. It is always good to have somebody look over your shoulder.

Mr. FAZIO. So your answer is yes, you think it would be?

Mr. PALLADINO. I think if you are talking about a technical group like the ACRS, I think that would be helpful, but they have to be aware of the whole mission and they have to weigh these factors.

Mr. FAZIO. They have to be aware of the overall mission of the agency.

Mr. PALLADINO. Yes, sir.

Mr. FAZIO. I am sure they would be.

Mr. PALLADINO. Those are my comments.

Mr. FAZIO. Anyone else?

Mr. ASSELSTINE. I think for myself since we haven't looked at those plants since 1974, I don't think there is a basis for our giving an opinion on how well or poorly they are run. I do have some concerns based upon apparently the previous concerns that the safety reviews identified back in the fifties, sixties and early seventies.

I want to go back and look at that information to see what specific concerns were raised about these confinement systems, about the adequacies of the seismic designs of the plants including the N-Reactor and some of these other factors. As far as the independent review is concerned, my personal preference would be for us to still focus our attention on the commercial reactors.

I think we have got our hands full there, but I also think there is a benefit in having an independent review to make sure that the military needs question isn't driving things to the point where safety gets sacrificed.

Mr. FAZIO. But you are not prepared to comment on the containment/confinement issue at this time?

Mr. ASSELSTINE. I would like to look at the documents. All I learned about them I learned last night and I will provide additional information for the record later.

Mr. FAZIO. Would these be available to the committee?

Mr. ASSELSTINE. I assume so.

Mr. **Fazio**. I would certainly hope we could have them for the record.

Mr. **Palladino**. May I correct a possible misimpression? When I said an independent review, I was not thinking NRC would review them. I was thinking just in general terms.

Mr. **Fazio**. Thank you.

Mr. **Bernthal**. I would just say, Mr. Fazio, I was somewhat surprised at the low level of communication that has existed between DOE and our agency in respect to exchange of thoughts and ideas about those particular production reactors.

We do have exchange of ideas and thoughts, as you probably know, in respect even to the Navy nuclear propulsion systems. But there apparently has been very, very little such exchange since 1974 for the production reactors as you have been told.

I have often said, when aspersions are cast to the contrary, that our level of communication with the Executive Branch, in fact, leaves something to be desired. There seems to be very little knowledge of what goes on in the NRC or what we even do much of the time.

Maybe this is one more example where we really should not stand so far away that there isn't even the benefit of our technical expertise.

Mr. **Fazio**. You are not going to the legal issue, but you are talking about the informal cooperation that would benefit everybody.

Mr. **Bernthal**. Not at all. There is a sensible way to do this. We are all concerned about our national defense and security. We are all also concerned about our public health and safety and I would hope there would be a common sense exchange of ideas between the two agencies.

Mr. **Zech**. Since the Navy program has been mentioned, let me just say it has been my experience that the Navy nuclear program is one of the finest programs our country has ever had. The record is certainly remarkable in all respects, certainly including safety.

I was very familiar at one time with the Navy's interface with the Department of Energy. It is my experience that the cooperation was superb. It was clearly a classy, professional organization run with the best interest of our country in mind and I was always extremely impressed with the Navy's part and with all the other parts of DOE that I was involved in from the weapons side, as well as the Navy propulsion side.

Now, in this new responsibility I have, in the commercial supervision of our commercial plants, I have focused my attention on the commercial side. That does not, though, change my impression and my confidence in what DOE is doing.

I have every confidence that they are doing a superb job. I have absolutely no reason to believe they are not doing just as good a job as we are and perhaps better.

So I think although it is always good to, in a circumstance like this, to review and take a step back and think carefully, I think that is appropriate for all of us to do that, I am sure that will be done between NRC and DOE and I can assure you that is my intention, to follow through on that.

Mr. **Fazio**. Thank you.

Harold, I hate to put you on the spot, but I would be interested in your thoughts. You have recommended that I get in touch with the DOE for help on my plant. So obviously you think there is some very capable people over there.

I would be interested in your thoughts particularly on the containment/confinement issue and the general evaluation.

Mr. **Denton**. My views on policy are no better than anyone else but it is an important topic. I would like to comment on your question.

I worked at Savannah River plant over 25 years ago and found continually they had large margins of conservatism in the design, very competent, large numbers of people at the plant, high levels of protection, but at that time the focus at Savannah River was on reducing the probability of an accident, whereas mitigating the consequences 25 years ago had not seeped into the consciousness of all of us.

Since that time they have gone to confinement. The technical point I want to make is that those plants are located on reservations with large distances between the plant and the nearest member of the public.

I am not really familiar with confinement they have in mind, per se, but often it is thought to be one in which you would release the noble gases but you would retain all the iodines and the particulates I think has been installed at Savannah River and the plants were not built with the idea of containment in size.

They are sprawling large and huge and it would be very difficult to confine them. I am sure that is why they have gone to a containment one which is to minimize the spread of particulate activity such as came out of Chernobyl and iodines and limit the release of anything to just noble gases.

Mr. **Fazio**. We have talked about a new facility rather than going back and retrofitting the existing facilities—It is almost impossible to retrofit the existing facilities.

Thank you, Mr. Chairman.

Mr. **Bevill**. The Chair recognizes Mr. Rudd.

Mr. **Rudd**. Thank you, Mr. Chairman.

Thank you, gentlemen. This committee has received testimony from you in the past indicating that the industry has taken a number of initiatives over the last few years to improve safety. As a matter of fact, you have pointed to the Palo Verde plant as an example of a high level of safety. As they have discovered better ways to do things in plant one, they save a lot of work and concern by factoring them into plants two and three. Perhaps the things that have been learned which as you have said has contributed to a high degree of safety, can be utilized throughout the industry. This could save time in terms of years and contribute to new areas that will allow the industry to grow.

Have I said anything there you disagree with?

Mr. **Palladino**. No, sir.

Mr. **Rudd**. Well, I just want to point that out, not only because it is something that is important to the entire nation, but also because it is in Arizona. I think that what you have said that the Palo Verde plants is something that has given a push to the industry and will help us to have safety in the future.

Now, everything I have heard here this morning tells me that we are a very safe and a high degree of safety has incurred in our industry. We should be proud of this. Yet I now hear something that disturbs me, because it seems like you are cautiously backing away from that and it all adds up to the fact you think you need $25 million more in your budget.

I hope that what you are saying is not detracting from the testimony we have had here this morning that tells me we have a safe industry. I think Commissioner Zech put it in a reasonable context by saying that if you did feel you needed additional funds, you would not hesitate to come forward and ask for them. That tells me that we don't have to back away from the testimony we have received. You gentlemen in your wisdom know that if you don't have enough funds you are going to come and ask us for them.

So I hope it has not detracted from the testimony we have had here this morning indicating that we do have a safe industry.

Mr. PALLADINO. Mr. Rudd, I don't think that was intended to detract from the safety of the industry, but it does point out that when events such as this take place you have to study them. You have to make sure there is nothing new or, if there is something new, that you give attention to it. That may take additional funds. Also, getting to where I think we were with regard to looking at the precursors to incidents, getting back to the capabilities we had then which I think pays dividends because it ends up in being——

Mr. RUDD. I think that is reasonable. I don't want to leave the impression that we don't have the high degree of safety in the industry.

Mr. PALLADINO. Incidentally, I also asked the EDO to check what he believes to be the resource requirements necessary to do the study on the Chernobyl reactor and what recommendations he would make.

Mr. BEVILL. Do you have a statement, Mr. Asselstine?

Mr. ASSELSTINE. One quick comment. I would say we have had a very successful program in this country thus far, but I do think there is a problem in uniformity of performance.

At some plants we are seeing a very high level of performance. In others we are not, and there is a large group in between. I would say we are not yet seeing the level of performance that is being achieved uniformly by some of the best nuclear programs in the world—I think basically Japan, West Germany and Sweden. Our plants, as a general rule, are not yet performing to the high level of performance of those plants. I think we ought to aspire to do that. There are some lessons we can learn from those programs in terms of care and attention in construction, in operation and in maintenance.

We ought to strive to do that and I think that will help ensure the kind of safety record we have had thus far will continue in the future.

Mr. RUDD. Commissioner Asselstine, I noticed your views here are pretty well stated in the morning paper.

Mr. ASSELSTINE. That is right.

Mr. RUDD. I am a little bit concerned about that. I was going to get to it a little later, but we might as well start on it now.

Again, everything I have heard here indicates we have a safe industry, but you have stated that we probably don't have a safe industry. If I were to read this without knowing better, I would think we are in a crisis position—that we have a dangerous industry. I will quote a couple things you have said. You have said that the Commission believes its job is to protect the industry and not the public.

That is frightening. And that there is an inexorable march down the path toward non-regulation of the nuclear industry.

There are a number of things you have said here.

The third thing you say, the memory of Three Mile Island has faded and complacency about safety has set in.

If those statements are true, then everything else that we have received in the way of testimony here this morning is not accurate. I don't know whether you want to expand on that or not.

Mr. **Asselstine**. I don't think that they are inconsistent at all, Mr. Rudd. I think those comments that I made, and I stand by all of them, really go to whether we can expect to see the same kind of success story in terms of safety for the future that we have had up until now.

I think it really goes to the question of whether we are satisfied with the level of safety and the level of performance being achieved by the plants today and whether we need to continue to pursue cost effective safety improvements in the program.

I think one of the messages from TMI was that there were some surprises there and that we needed to pursue cost effective safety improvements for the future and in particular we needed to focus on severe accidents to make sure that we don't see another one.

Now, the Commission has said, given the present level of the plants, we would expect to see another one, a 45 percent chance of another one in the next 20 years. In my view, that is not acceptable and we should not be satisfied with that level of performance and we should not be complacent about it.

We should be continuing to pursue actively safety improvements in the plants. I think our staff feels that way. Our staff recommended some of those safety improvements for Indian Point and the Commission rejected them.

Mr. **Rudd**. I am certain the staff feels that way. I am certain everybody feels that way.

We don't want to get complacent and we won't get complacent because everybody is working to be alert and they want to keep it that way and I think they ought to be commended for it.

Mr. **Palladino**. Mr. Rudd, I spent an hour with Mr. Weiskoff yesterday addressing the specifics of the arguments that have been set forth. I found it very discouraging that not one of my statements, except one statement that related to a severe accident, was quoted in the paper. So these notions do generate adverse opinion, as a matter of fact.

The writer displayed a point of view not only in what he was asking me to answer, but also in what he accepted in the answers I gave him.

Mr. **Rudd**. I think your answers were good and they were reasonable and they were on target, Mr. Chairman. I think you can summarize it by saying if we can't proceed on the premise our plants are safe, we should not license them.

I agree with that.

Mr. BERNTHAL. Mr. Rudd, I want to make a couple of comments.

One is you are absolutely right, you shouldn't go away with the impression that somehow $25 million is the objective here. It isn't.

I think most, if not all, of the Commission have said yes, we can live with the amount of money we have and it would be misleading to think that 25 more or less is somehow the straw that breaks the camel's back.

I do want to comment on a second point here and that is the Washington Post writing. In large, I think the Washington Post has not made a fair representation to say the least, of the efforts and the capabilities and the seriousness of this Commission and this body.

I was struck again on Sunday by the fact that in their lead story (one wonders how much professional oversight they have), they come forth with a statement that I believe that ran something like "the dose level was 300 R per hour, 200 times the lethal dose."

Well, that is just poppycock.

Mr. RUDD. I understand that.

Please forgive me, but in the interest of time, I did not criticize the article. I only stated things in the article. The words were taken from quotes and they were quoted in the article. I think I made that very clear. But I would like to ask some other questions, so if you have a comment to make on that, why don't you put it in the record?

Mr. BERNTHAL. If I can make a brief comment to the point then, I think there is no question that some of those quotations lend to the atmosphere that we had several events, for example, within the last year that somehow came very, very close to a Three Mile Island event. I doubt you would get that kind of opinion from our technical staff here.

That is not to say they are not serious events. We take these events seriously.

Mr. RUDD. I know you do. I want to bring out that you do take it seriously and there isn't complacency. Can you briefly tell me what procedures you have to warn the American people in the event of a nuclear mishap?

Mr. PALLADINO. We have quite an emergency response procedure that includes notification to the State and local governments and to other Federal agencies. They in turn have established procedures whereby the State and local governments through the use of sirens and radio network get the message to the public.

We also make sure that our International Program staff get the information to our neighbors in other countries.

Mr. RUDD. I will yield to my colleague.

Mr. MYERS. Along this line we do not have an elaborate civil defense system, but we do have a civil defense system. Are the civil defense participants trained by you as to what to do and how to act in such a case?

Mr. PALLADINO. Well, the State and local government people are participants in the planning. They work with the Federal Emergency Management Agency in developing training procedures in the offsite implementation of the evacuation plans or the shelter plans.

Mr. **Myers**. If the gentleman will continue to yield. Do they have someone close to these plants so they can go in and work quickly if we should have to? Do they have the training? The fact that they are in the plan doesn't mean anything if they don't have the training and equipment.

Mr. **Palladino**. We have the training, we have the equipment and back up of DOE teams. They have the equipment. We exercise the plan periodically onsite every year, off-site with the State and local government every two years.

We have also had major Federal response exercises that go beyond just the onsite response at a nuclear facility. So there is quite good support for these programs.

Mr. **Myers**. I thank you for yielding.

Mr. **Rudd**. Has there been loss of life as a result of radiation leaks in the history of the industry, Mr. Chairman?

Mr. **Palladino**. You mean to the public, non-workers?

Mr. **Rudd**. To anybody. I am talking about the nuclear industry as such.

Mr. **Zech**. Not to our commercial nuclear power industry, no.

Mr. **Rudd**. Thank you.

One other question. In the Comanche Peak plant in Texas, they are talking about computer software that will provide great assistance to operators in determining whether problems may arise from taking certain equipment out of service.

It seems this is probably an exciting effort. I wonder if you would comment on that?

Mr. **Palladino**. Let me ask my staff to comment on that.

Mr. **Denton**. I don't fully understand the question, Congressman, but I know that within the industry there are some exciting developments with application of computer programs to both diagnose and take corrective actions in the event they are needed.

Up to this point in time most of the actions in a plant have been what I would call by people in response to alarms and signals.

Mr. **Rudd**. This wouldn't relieve the operator of being thoroughly aware and cognizant of the regulation. I only use the plant in Texas because that is the only one I know about.

But continue.

Mr. **Denton**. Then there are attempts being made to use computers to assess what is the impact of taking an individual pump out of service. That sort of software is beginning to filter through the industry so it provides a good basis to say "if this pump is out of service then we better focus our attention on these remaining three batteries" sort of things.

So it can be used to better plan the day's maintenance activities and the corrective actions. But I would be happy to respond for the record in more detail on the Comanche Peak, if you like.

Mr. **Rudd**. I thank the committee, especially for the comments on Palo Verde and for your presence here.

Mr. Chairman, I have other questions, but I have overstated my time, so I will yield back.

Mr. **Bevill**. Chair recognizes Mr. Dicks.

Mr. **Dicks**. I wanted to go back and read this budget justification for DOE that I mentioned earlier. They said this project to meet the hangar replacement for seismic conditions will assure emergency core cooling and an elimination of core melt down during a seismic accident in which no direct fuel cooling is available.

The raw water supply systems of the graphite shield cooling system cannot now withstand seismic forces postulated for such an accident. Should such an accident occur, water may not be supplied to the system increasing the potential for core melt down. Is that a fair assessment of what the budget request is for?

Mr. **Hunter**. As stated there, that is essentially correct. We are talking about a seismic upgrade to the graphite shield cooling system. As I stated before, it is a totally distinct and separate system from the emergency core cooling system and from the normal reactor cooling system.

The reason that that was entered as a budget request developed from a hypothetical scenario in which, as I spoke of earlier, we talked about a partial core damage in the event that all normal cooling and all emergency cooling was lost.

In that scenario it was postulated that the graphite cooling system would remain in operation.

Mr. **Dicks**. Would you expand on that for the record?

A question for our friends from the NRC. Some members of the northwest delegation, not myself, have suggested that it would be a good idea for the NRC to conduct an independent review of the Hanford N-Reactor. If asked, could the NRC accomplish that mission?

Mr. **Palladino**. Nothing is impossible. We might be able to do it, but it would take getting completely familiar with the reactor, getting up to speed again on what the national security interests are and then, of course, taking the people and putting them on the job. I am not advocating it; I am just saying it could be done if that were what Congress desired. It would take resources that I don't think we have at the present time.

Mr. **Bernthal**. Mr. Dicks, I think the short answer is yes. I think the NRC could do that, but DOE would get a bill unless Congress made an appropriation.

Mr. **Dicks**. One final thing. You said you were cut from 460 to 405 million dollars. I would certainly like to see in the record what was cut out. Part of that cut was because of the first increment on Gramm-Rudman.

What would an $80 million cut from $405 million level mean? Mr. Asselstine, what affect, in your judgment would that have on the ability of the agency to carry out its safety related responsibilities? In other words, is Gramm-Rudman threatening your ability to protect the safety of the people of this country?

Mr. **Asselstine**. I think it is adversely affecting our ability now. It was about an $18 million reduction as I recall in the fiscal year 1986 budget, the current one we are in.

We did identify or I think we can readily identify what the impact of that is, which programs we have had to cut back on.

I think we were at a low level for 1986 already and the additional reduction with Gramm-Rudman does cut into our ability to do our job.

Mr. **Dicks**. Thank you, Mr. Chairman.

Mr. **Bevill**. Thank you.

Mrs. **Boggs**. May I ask one question, Mr. Chairman?

Mr. **Bevill**. Yes.

Mrs. **Boggs**. Thank you so much.

In 1982, there was a DOE computer study that postulated fatalities as high as 96,000 within a year if a reactor melt down occurred at the Waterford Three plant at Taft, Louisiana.

Could you please discuss the methodology of the study and the probability of this occurrence? Is there any reasonable expectation that this worst case scenario will occur?

Mr. **Palladino**. Mrs. Boggs, I would suggest we try to cope with that for the record if we may.

Mrs. **Boggs**. It is a question of some immediate importance, Mr. Chairman, because there have been stories in the New Orleans Times Picayune, with the second paragraph talking about the 96,000.

Mr. **Palladino**. We have to identify what stories we are addressing—what facts we have—and give you a response. If you feel something needs to be done within a matter of a few days, we will try to do it in a matter of a few days.

Mrs. **Boggs**. I would appreciate that very much.

Mr. **Palladino**. I think if you can give us a little more time in view of the many demands being made on our key people, it would be helpful. If you could live with two weeks, I suggest we get it to you within two weeks.

If you think it needs to be sooner, we will try to do it. Is two weeks all right?

Mrs. **Boggs**. I suppose so, if that is the time you need to have.

Mr. **Asselstine**. Mrs. Boggs, you may be referring to the Sandia study that was done by the Sandia Lab for us. If you are, my recollection was the assumptions that were made in that DOE study were worst meteorology in terms of dispersing the material and, I think, immediate early failure of the containment, the worst case assumption, so they would assume no protection from the containment and then the worst case weather conditions which led to those very high estimates of early fatalities or injuries.

I think the Commission testified at the time that some of those assumptions were very, very extreme in terms of the pessimism of them, in terms of failure of containment and the meteorology conditions, but we can check to make sure that is the study you are referring to.

Mrs. **Boggs**. This is the study. Not only was it referred to in the news story or front page, but also it is one of the coordinating activities between NRC and DOE that is certainly something we look forward to seeing a great deal more of.

I would very much appreciate some specifics on that.

Mr. **Asselstine**. Those were the worst case assumptions and they were for a number of plants where you saw very serious estimated health consequences, again, not taking into account the protection of the containment.

But I think if anything, what those things underscore is the benefit of making sure the containments will perform as expected and, second, the benefit of minimizing the chances of ever having a serious accident to start with.

Mrs. **Boggs**. Yes, Mr. Bernthal.

Mr. **Bernthal**. I just wanted to add a third. They also underscore what I hope we take away here today and that is the vital importance that we receive information and all countries receive information on causes, to be sure, of this accident, but also the effects of this accident. After the NRC has spent hundreds of millions of dollars, arguably billions of dollars—calculating and speculating, best scientific speculation on the scenarios for an accident, this may indeed be the bounding case for the worst case scenario.

We don't know until we get more information. It is terribly important, I think, to all of us to get that kind of information.

Mrs. **Boggs**. Thank you.

Mr. Denton?

Mr. **Denton**. I don't know the particular study, but of the hundreds of millions of dollars we have spent on research about how containments fail, we find it is very unlikely to have containment failure even in the event of severe core damage and if the containment does not fail the off-site consequences are nothing like those.

So they no doubt assume early containment failure.

Mrs. **Boggs**. Thank you.

Thank you, Mr. Chairman.

Mr. **Bevill**. Any other questions?

Mr. **Myers**. One observation, Mr. Chairman.

We thank our colleagues and witnesses here for being so patient today with all of us. This committee through the years has felt its responsibility to the American people not only for the safety of nuclear power but also the responsibility for providing the assurance that we have electric power. Today I think we are producing about 15 percent of our electric energy from nuclear and about 55 percent from coal with the rest coming from fuel oil, some hydro and some limited solar and geothermal.

We have legislation now being discussed across the street concerning acid rain legislation that will reduce the ability for us to use our own coal. If we can use it, it will be terribly more expensive. The Committee felt we needed to provide the American people the assurance that Congress is doing something about this terrible accident that happened in Chernobyl. We are concerned and are looking into the future to making sure it couldn't happen here.

We have often felt that the NRC has been somewhat negative in its approach towards its responsibility to provide for nuclear energy. We realize that you have been concerned about safety but often you have had the attitude of why you can't do something instead of how we are going to do it safely.

I felt that the American people were entitled to know something about what we were doing and the recent article in the Washington Post kind of destroyed that feeling. The American people are entitled to know that you were doing something about it and we collectively were doing something about it.

I am deeply concerned about this and this morning you talked about the need, I think Commissioner Asselstine, for 25 million additional dollars. Would you provide for the record what you would do with that money if the Commission got it?

Mr. **Asselstine**. Sure.

Mr. **Myers**. And what you are not going to do without it. We are facing a terrible responsibility here of cutting spending. We have to start living within our means somehow and there aren't any agencies or any segments of Government that are going to escape it.

Also, we have the responsibility to the American public to provide safe energy. We felt one of the safest and one of the cheapest was nuclear energy.

After the witnesses here, from both the Department of Energy as well as NRC, I think the evidence shows that it is still safe and we are doing everything we need to do but we want to make sure we weren't missing anything.

That is why we would like to know where that $25 million would be spent.

Mr. ASSELSTINE. Be glad to.

Mr. MYERS. Thank you very much for your testimony. It has been a long day, over three hours now, but it shows the concern all of us have over this incident.

Mr. BEVILL. Thank you. Thank each of you for coming.

I would like to say that there is one thing that has come out of this that I think is pretty obvious. All of us are trying to accomplish the same goal and there are some differences here on the $25 million. This Committee will certainly have to look into that, because if the Administration's recommendations is $25 million less than what is needed to maintain the safety of our nuclear industry this committee wants to know.

We will be glad to get your figures on that and what might be done with the $25 million. I think it is commendable that the Department of Energy has already started a study to see if there is anything that they may recommend to Congress to make the defense reactors handled by the Department of Energy more safe.

It has been suggested, of course, that it is always good to have someone looking over someone's shoulders.

My personal opinion is I think it would be a mistake to try to put the defense systems under the Nuclear Regulatory Commission. It is already under the Department of Energy and I think changing that would be a mistake.

You pointed out clearly we are dealing with two different types of reactors. The Secretary of Energy has already started his studies and, of course, the Department of Energy already has its Inspector General. If an additional Commission is needed, it may be logical.

I would like to get the comments from both you and the Secretary of Defense, because so far, I have seen no evidence to indicate any reason to try to place these facilities under the Nuclear Regulatory Commission which is dealing with an entirely different type system.

As Chairman Palladino pointed out, they would have to start off by making a study to understand the systems. I think that would be a waste of time and money.

Mr. MYERS. I think the Chairman has a comment.

Mr. PALLADINO. The only comment I wanted to make was the $25 million was not a Commission request.

Mr. BEVILL. I understand. Mr. Asselstine is the one that has requested that and he is going to explain that in his statement for the record and certainly this Committee is going to look at that.

Mr. MYERS. Please list your priorities, too. What is most important if we couldn't get all of $25 million.

Mr. ASSELSTINE. I will do that.

Mr. MYERS. And support why these are necessary.

Of course, I know you will do that. I have one other question.

The DOE has volunteered for a classified briefing. Would the NRC have anything that has not been discussed here?

Would you participate in the classified briefing?

Mr. PALLADINO. I am going to have Harold Denton comment. I have sat in on some of these briefings.

We in the NRC tend to work in the open and we prefer to work in the open. However, there may be pieces of information that we learn from sources that might be classified.

Mr. BEVILL. We will take that up in the Committee.

Did you want to comment?

Mr. DENTON. Other than to say that we, along with DOE, have access to the same information, therefore, their briefing should serve the purpose.

Mr. BEVILL. All right. The Committee will certainly take that up and consider that when we have a classified briefing.

Thank you for coming.

Mr. MYERS. You might also include where the $18 million reduction you took came from.

[Questions and answers for the record follow:]

Questions and Answers Prepared for the Record

REACTOR SAFETY

Mr. Bevill. Please describe the differences of the Department's production reactors to the reactor in the Soviet Union.

Adm. Foley. The Savannah River Plant (SRP) production reactors are a totally different design from that of the Soviet Union RBMK reactor at Chernobyl. The SRP reactors contain no graphite; no electric power is generated; they operate at low temperature (no boiling) and low pressure; and they employ advanced/redundant safety and control systems. The N Reactor at the Hanford site in Richland, Washington, is the Department's production reactor somewhat similar to the RBMK reactor at Chernobyl; however, there are important design differences between these two reactors which would make a serious accident at N Reactor very unlikely. The features of higher enrichment and uranium oxide fuel, online refueling, boiling coolant in the top third of the active core, and higher fuel burnup in the RBMK reactors make them more difficult to control and more susceptible to accidents. Although they have redundant cooling systems, the RBMK reactors have no comparable graphite cooling system equivalent to the N Reactor system, making them more susceptible to accidents initiating from multiple failures. N Reactor also has the following features which make a fire and release of radioactivity very unlikely:

o Graphite burns at greater 2,200°F. Under maximum operating conditions, the temperatures to which the graphite is exposed to range from 500°F at surfaces adjacent to cooling tubes to a normal maximum of 1,200°F in filler blocks.

- N Reactor core can be flooded with either an inert gas (helium or nitrogen) or water.
- A totally independent graphite cooling system exists at N Reactor.

In addition, N Reactor has the following multiple-redundant systems for which we have no confirmatory data regarding the Russian reactor:

- diesel-driven emergency core coolant pumps, two completely independent reactor shutdown systems, a graphite cooling system with redundant pumps and coolers, and two independent sources of electric power from offsite;
- a core coolant system with redundant and independent steam supplies to the primary coolant pumps;
- auxiliary electric power for those pumps; and
- a confinement system which provides barriers and controlled filtered releases to the environs.

Mr. Bevill. Compare the Department's production reactors to a typical civilian nuclear power reactor.

Adm. Foley. The primary purpose of the Department's production reactors is to produce plutonium and tritium for use in the Nation's defense, while in civilian reactors, the purpose is to produce steam to drive turbines to generate electricity. While the N Reactor does produce steam, it is only a by-product of the facility's primary mission of plutonium production. The Savannah River Plant reactors operate at low temperatures (210°F) and low pressure (5 psig) and do not produce steam. These low pressures and temperatures do not result in the large amounts of "stored energy" found in civilian nuclear power plants and thus, limit the forces that could be created during a reactor accident. The N Reactor confinement system design allows the critical venting of stored energy in the event of an accident to ensure the long-term operability of the confinement system.

Mr. Bevill. Describe the safety features of the production reactors and how these features eliminate the requirement for containment.

Adm. Foley. The Department of Energy employs a very conservative approach to ensure safe operation of its reactors. This philosophy includes the concepts of comprehensive and redundant monitoring to assure early detection of potential problems, shutting down its reactors before operating problems become serious and having multiple redundant engineered safety systems which are automatically activated if problems progress beyond present conditions. The success of this philosophy is evidenced by the fact that its currently operating production reactors have achieved a record of 127 years of reactor operation without an incident of significant consequences to onsite or offsite personnel. Each individual fuel/target assembly in DOE's production reactors is monitored to ensure that the assembly does not overheat and melt. As a matter of practice, the reactors are shut down instantly (scrammed) if any individual assembly shows signs of overheating or if any critical operating equipment needed to respond to an accident situation suddenly fails. The engineered safety systems are tested at regular intervals to ensure that they are capable of functioning when needed.

Each reactor has at least two independent safety systems which automatically shut the reactor down (before an operator can react manually) should the operating parameters (such as fuel assembly flow) exceed predefined limits. In the highly unlikely event that an accident does occur, there are independent, redundant sources of cooling to minimize damage to the core and subsequent release of fission products from the core. The confinement systems at each reactor are specifically engineered to provide continuous controlled, filtered release of airborne fission products from the building within established public health and safety standards. Each reactor has several other supporting engineered safety systems which are used to ensure the reactor operates within predefined limits, that the reactor will shut down automatically when required, and/or to mitigate the effect of any releases. The building housing the reactor and its safety systems is a massive reinforced concrete structure that was designed to be resistant to tornados, earthquakes, and blast effects. With these features, no credible accident results in damage to the building or to the capability of the safety systems to shut down the reactor and provide cooling for the reactor core. Radioactive materials released from accidents are retained by the confinement system, with the exception of the radioactive gases which are dispersed in a controlled manner to the atmosphere from a tall stack. Radiation exposure to offsite personnel is within the same guidelines established by the Nuclear Regulatory Commission for commercial reactors, even during the adverse atmospheric conditions. Full containment, is thus not considered necessary to cope with the highly unlikely accidents postulated for DOE reactors.

Mr. Bevill. From what we know or can assume from the accident in the Soviet Union, is it possible that containment would not have prevented the releases of radioactive material from that reactor?

Mr. Salgado. Until more definitive information is available on the accident and it's causes, it is extremely difficult to determine the impacts a U.S.-type containment structure might have had on the overall release of radioactive materials.

Mr. Bevill. What would it cost to provide containment to the DOE production reactors?

Mr. Salgado. The modification of the DOE production reactors from confinement to containment may be physically feasible but would clearly be an extremely difficult job costing approximately $5-6 billion and requiring 3-4 years of construction time.

REPLACEMENT PRODUCTION REACTOR

Mr. Bevill. Would a replacement reactor be a better alternative?

Mr. Salgado. The current use of engineered confinement systems for the Department of Energy (DOE) production reactors is safe, efficient, and adequate to protect the health and safety of the public. Containment systems are costly ($5-6 billion to outfit all five production reactors) and would not necessarily enhance the safety associated with existing design. In order to replace the current nuclear materials production capability of the five production reactors currently available, at least two and probably three new production reactors would have to be built depending on the final selection of reactor type to be used. The final cost for this effort could reach $10-13 billion depending on the number of reactors built, type of reactors built, and on the site selected to build the reactors.

Mr. Bevill. What is the expected cost of a replacement production reactor?

Mr. Salgado. The estimated cost of a new large production reactor would range from approximately $2.8 billion for a Heavy Water Reactor to $4.4 billion for a High Temperature Gas Reactor in 1985 constant dollars.

Mr. Bevill. How long would it take to bring on line?

Mr. Salgado. It is estimated that it would take approximately 12 years for design, construction, and preoperational testing to bring a new production reactor on line.

Mr. Bevill. What would be the impact on the nuclear weapons program if the production reactors were shut down and replacement reactors were to be built?

Mr. Salgado. The nation's nuclear deterrent is critically dependent upon the operation of the existing defense production reactors for special nuclear materials. An outage of all defense reactors equivalent to the construction period required for new production reactors to be built would cause a massive reduction in the effectiveness the nation's nuclear weapons stockpile as a deterrent force.

N REACTOR

Mr. Bevill. What would be the impact on the nuclear weapons program if the N Reactor in Richland were to be shut down?

Mr. Salgado. The N Reactor at the Hanford site makes a substantial contribution to the material requirements of the nuclear weapons stockpile. Its absence would seriously impair the stockpile modernization program which is designed to enhance the safety and security of the United States nuclear weapons compliment.

Mr. Bevill. From what we know about the nuclear accident in Russia, what advice do you have for this Committee and the American people?

Mr. Salgado. The accident at Chernobyl is an extremely serious event. It should remind all of us of the grave responsibilities we bear and strengthen our resolve to continue safe, careful operation and oversight of our nuclear reactors. We also must learn all we can about the Chernobyl accident and apply any lessons. However, this process must be deliberate and reasoned, and any actions must be taken judiciously.

Mr. Bevill. Please describe the DOE program to extend the life of the "N" Reactor.

Adm. Foley. Three specific programs have been developed over the last 6 years for continued operation of the N Reactor. They are restoration, productivity retention, and production assurance. The first program was necessary to regain operational performance after some years of minimal funding and is essentially completed. The other two programs are focused on identifying life-limiting conditions and then preparing surveillance procedures, enhanced maintenance programs, and upgrade plans responding to these conditions before safety and production minimum levels are reached. These programs have been further described in budget submittals.

REACTOR SAFETY

Mr. Bevill. What plans does the Department have with regard to building a new replacement production reactor?

Mr. Salgado. The decision to deploy a new production reactor has been deferred for several years until completion of life studies for the existing Savannah River Plant production reactors and refurbishment plans for the N Reactor are completed and requirements for materials production confirm the need for a new production reactor.

DOE SAFETY REVIEWS

Mr. Bevill. Please describe what additional safety reviews have been initiated as a result of the accident in Chernobyl.

Mr. Salgado. Prior to the Chernobyl incident, DOE had scheduled Technical Safety Appraisals at the five production reactors. They are the N Reactor in Hanford, Washington, and four reactors at the Savannah River site in Aiken, South Carolina. In response to the Chernobyl accident, the Secretary directed that these schedules be accelerated. In addition he ordered two other reviews as a result of issues raised by the Soviet incident. First, a Special Safety Team began May 5 to review key aspects of N Reactor safety including confinement design and operation and graphite considerations. (Other than the use of graphite, the N Reactor bears little similarity to the reactor at Chernobyl). Second, starting this week, a separate Design Review Team will perform a detailed review of the N Reactor design. This is the fifth such design review for N Reactor since 1963. The accelerated Technical Safety Appraisal, one of Secretary Herrington's 1985 ES&H initiatives, will now begin the week of May 19, 1986. The Technical Safety Appraisal of the Savannah River reactors will begin in September rather than as previously scheduled in December.

The Technical Safety Appraisals will review all aspects of safety, including among other areas, operations, emergency planning, and maintenance. The DOE ES&H independent review teams will be made up of experienced DOE technical experts and recognized experts from the Department's national laboratories and private industry. For example, the Special Safety Review Team which started its effort on May 5th, includes technical experts in reactor safety, graphite reactors, reactor containment/confinement systems, and graphite fire safety. The two ES&H reviews that follow the Special Safety review, the Design Review and Technical Safety Appraisal, similarly will utilize both DOE and outside technical specialists and experts. Secretary Herrington has also requested several nationally recognized scientific and technical experts to review the safety of the Department's N-Reactor and to provide him with their views.

In a broader context, the Secretary has requested the National Academy of Sciences and the National Academy of Engineering to make an independent asessment of the Department's 11 major production and research reactors. This assessment is to make certain that we have thoroughly examined all aspects of the nuclear safety issue in light of the Soviet accident.

GRAPHITE CHARACTERISTICS

Mr. Bevill. If graphite is made the culprit at Chernobyl, what evidence do you have that the graphite burned? What contribution did decay heat, the zircaloy-water reaction, and graphite make to the heat generation following the accident?

Mr. Vaughan. A summary description of the Chernobyl accident made available by the International Atomic Energy Agency at the conclusion of the visit by its Director General to the Soviet Union refers to a graphite fire. While it would seem probable that a zircaloy-water reaction occurred, the contribution of this chemical reaction, its hydrogen reaction product, and decay heat to the heat generation required to either initiate or sustain the fire is uncertain. This is particularly so in light of the breach in the Unit 4 reactor building, which exposed the graphite to the environment.

CHERNOBYL ACCIDENT

Mr. Myers. Do we know for certain when the accident occurred? What is the most current information on its condition now?

Mr. Vaughan. The Soviet government reported that an explosion occurred at the reactor site at 1:23 a.m. on April 26, 1986. It is quite possible that initiating events began several hours prior to that time.

At the present time, it appears that the reactor has released a significant amount of its fission product inventory with radiation levels now decreasing. The condition of the reactor core is not known at this time.

TRAVEL ADVISORIES

Mr. Myers. Is DOE, NRC, or the State Department advising Americans to avoid travel to any countries that may have been effected by this event?

Mr. Vaughan. Other than a travel advisory for regions in the Soviet Union near the accident area, no advisories have been issued to avoid travel elsewhere.

DOE BUDGET REQUEST

Mr. Myers. Suppose you were marking up this bill today. Are there any changes to your own budget request that you would consider as a result of the Soviet reactor?

Mr. Salgado. I would make no changes to our budget request based on the currently limited information available relative to the accident at the Soviet reactor. I believe our current budget request provides for a program of safe and efficient operation of those facilities necessary to meet the requirements for the Department's missions. Should future information from the Soviet accident or any other analysis indicate a need for funding to continue the safe and efficient operation of these facilities, the appropriate funding request will be submitted.

INTERNATIONAL DISCUSSIONS

Mr. Myers. In light of the international overtones of this event, is there any significant discussion among governments about international inspection or even licensing of nuclear plants?

Mr. Vaughan. At this point in time, we are not aware of any significant intergovernmental discussions regarding international inspection or licensing of nuclear plants. We would expect that the International Atomic Energy Agency would serve as a forum for such discussions. I would note that the IAEA plans to convene an extraordinary Board of Governors meeting in Vienna on May 20 to discuss the accident and possible follow-on Agency actions.

N-REACTOR DESIGN

Mrs. Boggs. Is the Hanford N Reactor closer in design to the Chernobyl plant or to the typical U.S. commercial reactor?

Adm. Foley. While N Reactor has some features similar to the Chernobyl plant, such as a graphite moderator, and other features similar to a U.S. commercial power plant, such as a defense in-depth design approach and multipli-redundant safety systems, it would be inappropriate to characterize it as being like either type of plant.

Mrs. Boggs. Would you characterize the differences between the Chernobyl reactors and the N Reactor as fundamental differences in design philosophy or as differences in engineering detail?

Adm. Foley. There are fundamental differences between the Chernobyl reactors and the N Reactor. The Chernobyl site has the Soviet RBMK-1000 reactors which are designed for electricity generation. The N-Reactor is designed for materials production. For example, N Reactor fuel is in a horizontal configuration and is cooled by pressurized water; this configuration permits efficient and safe production of defense nuclear materials. The Chernobyl reactor fuel is in a vertical configuration, and is cooled by boiling water; a design which enhances steam collection for electricity production. More importantly, Chernobyl may not be have the degree of safety redundancies that are used on N Reactor to protect the public. For example, the Emergency Cooling System on N Reactor provides for system depressurization and reactor cooling even in the event of loss of two offsite power sources and loss of the onsite backup electrical supply provided by a turbine/generator. Loss of this backup is extremely unlikely as it has the three independent sources of steam.

Mrs. Boggs. Many experts believe that the accident at Chernobyl became severe because of an explosion of hydrogen gas produced during the early stages of the accident. A hydrogen explosion occurred at Three Mile Island, but it did not have serious consequences because of the strength of the containment building. The NRC has required U.S. reactors with small containment buildings to take extra precautions against hydrogen explosions. Is it possible for hydrogen to be produced at the Hanford and Savannah River plants?

Adm. Foley. The production of small amounts of hydrogen are possible during hypothetical accident situations in both the N Reactor and the Savannah River Plant (SRP) reactor. However, the confinement system is designed to prevent the buildup of explosive gases, including hydrogen, in the reactor buildup.

Mrs. Boggs. How do the amounts of radioactivity at the Hanford N-Reactor and the reactors at Savannah River compare to the amount of radioactivity in the average U.S. commercial reactor? And how does it compare to the Chernobyl reactor?

Adm. Foley. The Department of Energy's production reactors at the Savannah River Plant and the Hanford site are designed and operated with the primary mission of producing defense nuclear materials. The N Reactor at the Hanford site, however, does produce steam as a byproduct for generating electricity. The operating cycle for production reactors; i.e., residence time of uranium fuel or targets in the reactor core, is much shorter than the uranium fuel in the core of a power producing reactor. For example, the fuel burn-up is about 1,000 megawatt-days per metric ton (MWD/MT) for N Reactor, about 20,000-35,000 MWD/MT for a typical U.S. commercial power reactor, and about 18,000 MWD/MT for a Chernobyl reactor. The amount of radioactivity in the core of a production reactor is considerably different than that in a power generating reactor. The quantity of short-lived radioactivity is comparable for all reactor types because they operate at approximately the same power level. However, the longer fuel residence at a Chernobyl-type of reactor, or of a commercial nuclear power plant results in a considerably greater quantity of long-lived radioactivity.

REACTOR SAFETY

Mrs. Boggs. What safety precautions are taken against hydrogen explosions?

Adm. Foley. The confinement system prevents the buildup of hydrogen in the reactor building. The scenarios evaluated for DOE production reactors indicated that hydrogen concentrations will be no greater than 1.2 percent which is well below the 4 percent concentration needed to burn or the 5.7 percent concentration needed in air for an explosion.

Mrs. Boggs. Would the amount of radioactivity that might escape from a release at the Hanford reactor or the Savannah River reactors in the worst case be substantially different from what is believed to have escaped at Chernobyl?

Adm. Foley. Based on the safety analyses for the Department's production reactors, the amount of radioactivity released in the postulated worst credible accidents would be very small in comparison with Chernobyl. As a matter of fact, the resultant doses to the public for all of our production reactors due to a postulated maximum credible accident would be a small fraction of the dose guidance given in the Nuclear Regulatory Commission's reactor siting criteria (10 CFR 100).

Mrs. Boggs. I understand that one of the safety advantages of the Savannah River reactor is their low operating temperature (about 115°C). That compares to much higher temperatures for the RBMK reactor at Chernobyl and commercial pressurized water reactors in the U.S. How long would it take one of the Savannah River reactors to heat up to the operating temperature of a commercial reactor if its cooling capacity were lost, for example, if the core were uncovered?

Adm. Foley. In the situation you hypothesize of loss of the primary coolant and the four sources of emergency core coolant and that the core of a Savannah River Plant reactor uncovering completely, the fuel would melt quickly and the fission products would be released into the reactor building and its confinement system. The low operating temperature and pressure which contribute to relatively low stored energy, are some of the primary reasons that damage to the core is not considered a credible accident. Compared to commercial reactors, the low operating temperature and pressure of the SRP reactors is a safety advantage from two standpoints. The first is that if the primary coolant is low (for example due to a large break in the primary piping), the supplies of emergency cooling water have sufficient pressure to enter the core to preclude damage to the assemblies. Secondly, is a hypothetical large break in the primary cooling system does not result in enough steam formation to overpressurize the building or the confinement system.

REACTOR SAFETY

Mrs. Boggs. Why is that extra time a major safety advantage?

Adm. Foley. Upon loss of normal cooling water by a commercial nuclear plant, the time for the core to heat up is very short; conversely in the case of the SRP reactors, which operate at low pressure and temperature, the core will heat up more slowly upon loss of primary coolant. In both cases, emergency cooling water systems would come into play to preclude damage to the fuel assemblies.

Mrs. Boggs. How long would it take before the fuel would be damaged if the core were uncovered?

Adm. Foley. If the core of a Savannah River reactor were uncovered as a result of a hypothetical event, some fuel could begin to melt within a minute. However, one of the major safety advantages of the low operating temperatures and pressures of the Savannah River Plant reactors is that upon loss of primary coolant, the emergency cooling water supplies will preclude damage to the fuel assemblies.

Mrs. Boggs. What would be the consequences of such an accident?

Adm. Foley. The worst credible loss of coolant accident would result in the release of 1% of the core inventory of radionuclides. The release of radioactivity to the environment, however, would result in doses within the NRC reactor siting guidelines given in 10 CFR 100.

Mrs. Boggs. Given the danger posed by the lack of a confinement structure at the Chernobyl plant, are you confident that a core damage accident is so unlikely at Hanford or Savannah River that containment buildings are not necessary on the reactors there?

Mr. Salgado. Yes. Due to the design of the Department of Energy's production reactors and their engineered safety systems, the probability of accidents which would result in some limited core damage is remote. These accident scenarios have been extensively analyzed and the design of the confinement systems were developed to keep offsite releases of fission products, even during the worst conceivable atmospheric conditions, within the limits established by 10 CFR 100.

Mrs. Boggs. On the basis of about two dozen probabilistic risk assessments (PRAs) for licensed nuclear power plants in the U.S., the Nuclear Regulatory Commission staff estimates that the frequency of a core damage accident is from one in 1,000 years to one in 10,000 years. That risk is made acceptable by the presence of containment building which are expected to greatly reduce the consequences of the large majority of core damage accidents. Have there been PRAs performed for the Hanford and Savannah River Reactors? If not, why not?

Adm. Foley. PRAs must be both plant and site specific to properly account for unique characteristics of each type of reactor, its safety systems, and its siting in relation to the general population. An N Reactor PRA was initiated at the beginning of FY 1986 and will be completed in FY 1988. The PRA for the Savannah River Plant reactors was initiated in FY 1985 and will be completed in FY 1988.

REACTOR SAFETY

Mrs. Boggs. Do you think the risk of a core damage accident at one of these plants is acceptable?

Adm. Foley. The design, operation, and maintenance of the Department's production reactors at the Savannah River Plant and the Hanford site use the "defense in-depth" concept. The multiple-redundant reactor safety systems would contribute to and assure the extremely low probability of any accident that could result in core damage. Current safety analyses for these production reactors show the potential consequences are within established standards and the risk; that is, the product of probability and consequences, is acceptable.

Mrs. Boggs. Do you have good reason to believe the frequency of core melt accidents at those reactors would be less than at licensed U.S. facilities?

Adm. Foley. The safety analyses for the production reactors show that many safety features included on the plants afforded a high degree of confidence that the probability of a core melt accident is extremely small. The technology of quantitative risk assessments is not such that we can make a direct comparison with licensed commercial power reactors.

Mrs. Boggs. Have you identified scenarios that contribute the most to the risk at Hanford and at Savannah River? If so, what are those scenarios?

Adm. Foley. The postulated accidents which result in the greatest radiation exposure to the public are identifed in the N Reactor Updated Safety Report as a flow blockage in a single pressure tube, which could lead to fuel melting in that pressure tube, or complete end cap failure in a single pressure tube which would eject fuel from the tube and is postulated to lead to

fuel melting. The consequences in terms of doses at the low population zone are much less than the siting guidelines given in 10 CFR 100.
For the SRP reactor, the postulated accident which resulted in the greatest radiation exposure is a localized nuclear criticality resulting from a misloading of fuel. Core damage in this accident is less than 3 percent and the maximum resultant whole-body and thyroid offsite dose would be within the dose guidelines given in 10 CFR 100. The probability of either accident occurring is about one in a million years.

Mrs. Boggs. How large a risk do they represent?

Adm. Foley. The consequences in terms of the maximum resultant whole-body and thyroid doses at the site boundary are much less than the siting guidelines given in 10 CFR 100.

Given the low probability of an accident of this magnitude and the availability of systems to control any accidental releases of fisson products at either the N Reactor or the SRP reactors, the risks are extremely small.

EMERGENCY RESPONSE PLANS

Mrs. Boggs. What type of emergency response plans are there for the population around the Hanford and Savannah River reactor sites?

Mr. Salgado. Both the Hanford and Savannah River sites of the Department of Energy (DOE) maintain comprehensive emergency preparedness plans and procedures to support onsite and offsite emergency response actions in the event of an accident. These plans and procedures are totally integrated with those of offsite authorities such as State and local governments. They reflect conservative estimates of the consequences of the most severe credible accident at each nuclear facility on these sites. A major exercise is conducted annually at both Hanford and Savannah River, frequent drills are conducted at each facility, and all personnel are required to participate in the emergency preparedness training programs. In addition, emergency prepardeness is included as a topic in the extensive DOE site and facility safety appraisal programs.

EMERGENCY RESPONSE PLANS

Mrs. Boggs. How do they compare with offsite emergency response plans required at commercial reactor sites?

Mr. Salgado. The emergency preparedness programs at DOE sites, particularly at both Savannah River and Hanford, are comparable to those of commercial reactor sites. Almost all of the emergency planning zones (EPZ) for the DOE facilities at Savannah River and Hanford are within the site boundary; thus, the DOE field office managers, who report directly to the Under Secretary of Energy, are in control of all persons within the EPZs. They also maintain close relationships with State and local officials, and with regional Nuclear Regulatory Commission and Federal Emergency Management Agency offices, to assure the precautionary offsite actions will be effective if they are needed in the event of an accident.

EMERGENCY RESPONSE PLANS

Mrs. Boggs. Given the lack of containment buildings, how would the time available for evacuation or other emergency response measures compare for these reactors and for licensed reactors?

Mr. Salgado. The time available for evacuation or other emergency response measures depends on many things including meteorological conditions and distance. The notification process at the sites where the Department's production reactors are located have been coordinated with South Carolina and Washington. Because of the large distances between the DOE production reactors and the site boundary, the notification times are adequate.

CHERNOBYL REACTOR

Mr. AuCoin. How long had the Chernobyl reactor been in operation?

Mr. Vaughan. An April 17, 1985, news release from the Atomic Industrial Forum cites a 1983 operation date for Unit 4 at the Chernobyl complex.

Mr. AuCoin. What is the Chernobyl reactor's design lifetime?

Mr. Vaughan. We believe the design lifetime of the Chernobyl reactor is 30 years.

N REACTOR

Mr. AuCoin. How long has the Hanford N-Reactor been in operation?

Adm. Foley. Full power operation began at the end of 1964, and startup tests began a year before that.

SOVIET EMERGENCY PLANS

Mr. AuCoin. What were the Soviet's pre-existing plans--if it can be determined there were any--for dealing with a serious accident at Chernobyl?

Mr. Vaughan. We have no information at this time on the Soviet's emergency plans for dealing with a serious accident at Chernobyl.

Mr. AuCoin. Were there evacuation plans in the event of such an accident?

Mr. Vaughan. We are unaware if there were evacuation plans in existence at the Chernobyl site prior to the accident.

N-REACTOR

Mr. AuCoin. What is the Hanford N-Reactor's design lifetime?

Adm. Foley. For particular components where usage was important, the design value was 25 years of operating time with maximum use of the component in question. After reactor operation began, this value became inappropriate in determining the reactor's lifetime because the lifetime is dependent on the actual condition and observed performance of the plant. The N Reactor maintenance and upgrade programs are designed to identify and replace or repair components as they approach their "end of life".

Mr. AuCoin. If N-Reactor experienced an accident similar to the one at Chernobyl, (1) what would be the human health effects? (2) what would be the environmental effects? (3) how and when would the public be notified?

Adm. Foley. We don't know enough about the accident at Chernobyl to speculate about what would be the human health effects. We can state that analysis of all credible and a maximum hypothetical accident for N Reactor shows compliance with 10 CFR 100 dose guidelines, and would have minimum human health effects. Potential environmental effects are also addressed in the N Reactor updated safety analysis reports.

N Reactor offsite evacuation planning is documented in the Richland Operations Office (RL) Emergency Plan, dated January 11, 1982.

A letter of agreement between RL and Grant County has been in effect since April 7, 1982. This Agreement outlines roles and responsibilities for evacuation. A letter of Agreement between the Wahluke School District and the State of Washington, dated March 5, 1982, provides for the Mattawa Public School to be used as a reception center for evacuees.

A door-to-door survey and public instruction program was completed by RL, UNC Nuclear Industries, and Grant County Department of Emergency Management on March 12, 1986, to all the residents living within 10 miles of N Reactor. Specific evacuation instructions were provided--a 1986 Emergency Calendar which lists all the actions to be taken in the event of an emergency was left with each resident. Also notification will be provided via a computerized notification system with all residents telephone numbers put into the system. (There are only 35 permanent residences and businesses within the N Reactor contingency emergency planning zone). The testing of the system is ongoing and was most recently tested on April 7, 1986, and May 5, 1986. Also, formal agreements are in place to provide backup notification via the emergency broadcast radio system.

A formalized system is in place to notify the people on the Columbia River in the event of an emergency at N Reactor. A memorandum of understanding among RL, Coast Guard and the County was signed May 21, 1982. The Columbia River Evacuation plan was last tested on August 12, 1985.

N REACTOR

Mr. AuCoin. As a reactor approaches retirement age, does it become safer or more accident prone? Why?

Adm. Foley. The DOE production reactors are all constantly maintained to ensure that there is no decrease in safety.

Mr. AuCoin. What problems does the Department expect will be associated with N-Reactor as it reaches the end of its expect life?

Adm. Foley. No foreseeable technical or operational problems, outside of known engineering solutions, are expected because programs have been established to account to wear and cyclic operation through either continued maintenance or replacement. One or two extended shutdown periods will be required to incorporate these upgrades.

Mr. AuCoin. Have any of these problems been observed yet?

Adm. Foley. The technical and operational issues deal with maintainability and operational costs associated with older technology components and with time dependent changes of components such as wear or corrosion that are associated with usage. Two particular conditions that will require future replacement or rework are the reactor's pressure tubes and graphite stack. Enhanced surveillance of these latter components show modification or replacement projects will be necessary in the mid-1990's. The extensive maintenance program at N Reactor (about $50 million per year) and the programs described earlier are identifying and correcting components that are reaching "end of life".

Mr. AuCoin. According to an article in The Oregonian dated May 4, 1986, a DOE official and an employee of the firm which operates N-Reactor said "the worst accident contemplated at the plant would involve the meltdown of fuel in just one of the process tubes." How many process tubes are there?

Adm. Foley. The N Reactor contains 1,003 horizontal pressure tubes.

N REACTOR

Mr. AuCoin. What condition are the tubes in?

Adm. Foley. The material properties and dimensional characteristics are being monitored constantly. As of today, the large majority of these tubes are expected to meet the required code strength an flexibility properties until at least the mid-1990's. Some maintenance activities will be required for the balance before then. At that time, program plans will be fulfilled to replace all the tubes. In addition, all service- and environment-induced conditions of the tubes are normal and still meet the original design goals.

Mr. AuCoin. As they age, don't the tubes get distorted? Isn't that dangerous?

Adm. Foley. The pressure tubes are not as straight now as they were when first installed because of dimensional changes of the moderator surrounding the tubes. The effect, however, is small and typically is only about 1 1/2 inches in 50 feet or less than 0.3 percent. Since the dimensional changes an the tube material properties are known, it is assured that the tubes are still operating within the allowable code.

CONTAINMENT VERSUS CONFINEMENT

Mr. AuCoin. Why does the NRC require containment structures for commercial reactors, rather than confinement structures?

Mr. Vaughan. One of the original motivations for containment was to provide a truly passive, leak-type barrier so that commercial powerplants might be located closer to population centers as compared with remotely-sited DOE facilities. The purpose of a containment structure is two-fold: first, to control pressure and, second, to control the release of fission products into the environment.

Light water reactors licensed by the Nuclear Regulatory Commission operate at pressures well above 1,000 pounds per square inch. In the event of a major leak or break in reactor coolant piping, water that escapes will flash into steam. Containment systems have been the traditional means of assuring that the pressures resulting from the hot steam are controlled.

Containment structures also constitute a final barrier to protect the public from an accidental release of radioactivity.

A combination of containment and water sprays and filters are generally used in commercial reactors to reduce the consequences of accidents to acceptable levels. Confinement structures perform the same function but allow controlled venting of noble gases. For reactors located close to population centers, confinement systems may not afford the same amount of time to notify the public and take protective actions on their behalf. Noble gases are generally much less hazardous than other fission products that can be released from a damaged core. A combination of confinement and distance from the public can be a very effective means of assuring public safety but the NRC, with few exceptions, has not permitted their use in the commercial reactors.

Mr. AuCoin. What is the difference between a containment structure and a confinement structure?

Adm. Foley. U.S. reactor safety utilizes a "defense in-depth" principle throughout reactor design, operation, and maintenance to meet established standards. Defense in-depth is accomplished by installing numerous safety systems to prevent or control the release of radioactivity.

The terms "containment" and "confinement" are commonly used in referring to systems used at nuclear reactors to reduce the quantity of radioactive material that might be released to the environment in a maximum credible accident. "Containment" and "confinement" are just two of the many, engineered safety systems which contribute to this "defense in-depth" concept.

Containment involves "essentially a leak-tight structure" which permits periodic safe release of radioactive gases. The structures typically consist of reinforced concrete up to several feet thick.

Confinement allows for filtration and removal of airborne radioactive materials with a continuous safe release of radioactive gases within all applicable federal public health and safety standards. The structure can consist of up to several feet of reinforced concrete. N Reactor and the Savannah River reactors are examples and utilized thick concrete reinforced walls.

Mr. AuCoin. How much pressure can a containment structure withstand?

Adm. Foley. Each containment structure has a certain pressure retention capability based on the specific design. For example, large dry containments in the United States are designed for about 40-50 psi pressure to withstand the pressurization due to a complete rupture of the largest pipe in the system. The high design pressures are necessary because of the very high energy inherent in pressurized water and boiling water power reactors.

Mr. AuCoin. How much can a confinement structure withstand?

Adm. Foley. Because any fuel failure or release of radioactive material would not occur until some time after the depressurization and blowdown of the N Reactor primary loop, the N Reactor confinement system includes a provision for venting the steam from the blowdown during the first few minutes of the event. Then the vents are closed and a controlled, filtered release established. The N Reactor confinement building is designed to withstand a 5 psig overpressure or a 2 psig underpressure. The design overpressure is much lower than in commercial power plants because of the lower residual stored energy in the reactor system.

The Savannah River Plant reactor confinement systems are not designed for an internal pressurize because even in the most severe accident situation, the reactor has little stored energy due to low operating temperature and pressure. However, the SRP reactor buildings are designed to withstand a 1,000-pound per foot overpressure from the outside.

Mr. AuCoin. Would a containment structure be better to have in case of a fire or explosion at N-Reactor?

Mr. Salgado. The confinement system in use for N Reactor is like a low pressure containment in that early high pressure clean gas is dissipated through vents so that N Reactor's radioactive barrier will operate at very low pressure compared to the environment. This design has been demonstrated to meet established standards.

The N Reactor confinement system is designed to prevent the buildup of concentrations of explosive gases in the reactor building.

N REACTOR

Mr. AuCoin. Has there ever been a fire at N-Reactor? If so, how many have there been? What were the consequences?

Adm. Foley. Within the confinement area there have been no reported fires since June 21, 1967. On June 21, 1967, there was a major fire in Cell #3 of the 109-N Building. It was a hydraulic oil fire on the primary coolant pump. Consequently, the reactor was shutdown for a short period of time. The fire caused major damage to the equipment around the Primary Coolant Pump. The reactor was safely shutdown with no release to the environment. As a result of that fire, major fire protection features were added for all Primary Coolant Pumps. These features included: Installation of automatic sprinkler protection around each pump; a hydraulic oil containment system around the pumps; and increased surveillance of hydraulic equipment for the Primary Coolant Pumps.

Fires resulting in more than $1,000 damage are required to be reported.

Mr. AuCoin. What would be the prospects of release of radioactive material into the Columbia River in the event of an accident at N-Reactor?

Adm. Foley. All significant releases of radioactive material directly to the environment from N Reactor would be through the confinement system and the stack. None would be directly released to the Columbia River even in an accident situation. Any liquids would first be placed in a large tank, with the majority of the released radioactivity to be vented and disposed of later.

Should the tank capacity be insufficient, slightly contaminated liquid might be released to a "crib" and the soil column used to further reduce the radioactive content. The doses calculated for the postulated accidents consider these factors and they are within applicable standards.

WASTE DISPOSAL

Mr. AuCoin. Does the Savannah River facility use seepage bins for disposal of radioactive waste?

Adm. Foley. Seepage bins, or basins, are in use at Savannah River for disposal of radioactive waste. The contaminated waste discharged to these basins contains very low levels of radioactivity. Operations with radioactive materials are contained within the 300 square mile Federal site. Industrial scale operations are conducted with large quantities of radioactive materials in the national defense interest. A wide buffer zone provides physical security and assures that concentrations of radioactive materials in air and water will be extremely small at the plant boundary. Any ground water contaminations is within the site boundaries and does not create a hazard to the public or to employees.

Mr. AuCoin. How much longer will those seepage bins be in use at Savannah River?

Adm. Foley. In 1983, the U.S. Department of Energy's Ground Water Protection Plan for the Savannah River Plant was submitted to the Congress. This Plan addressed the remedial action and closure of seepage basins serving three areas of the plant that had the major impact on the ground water. For one area, a treatment facility has been built and seepage basin operations were discontinued in 1985. For the other two areas, a treatment facility is scheduled to begin operation in 1988, at which time seepage basin discharges will be discontinued.

No decisions have been made on the continued use of seepage basins at Savannah River Plant for disposal of low level radioactive waste. An environmental impact statement (EIS) is being prepared to assess the potential environmental effects of modifying waste management activities for low-level radioactive, hazardous, and mixed waste on the protection of human health, ground water, and the environment. In this process, the continued use of these seepage basins will be assessed.

Remedial and closure actions for all seepage basins will be determined to comply with applicable regulatory requirements, including the Resource Conservation and Recovery Act. Specific remedial and closure actions to be taken will be established by the EIS record of decision planned for August 1987.

Mr. AuCoin. Are seepage bins being used at the Hanford facility?

Adm. Foley. There are currently 30 seepage basins in operation at Hanford. Cooling water is discharged to engineered filtration basins called ponds. Low-level liquid waste is dischaged to engineered basins called cribs. A crib is an underground disposal system similar to a drain field.

Seepage basins will be needed to support production operations until feasible alternative disposal methods are developed. A study is nearing completion that considers alternatives to the present disposal practices in terms of health and safety effects, cost, and impact on production operations.

Mr. AuCoin. Is the Department planning to construct additional seepage bins?

Adm. Foley. The FY 1987 President's budget identifies two candidate general plant projects to construct ponds for receipt and disposal of normally uncontaminated water from facilities which operate in support of DOE's nuclear materials production mission. These will be required in order to assure continued safety of existing facilities and operations.

The FY 1987 GPP request was based on estimates of facility requirements at the time of preparation of the FY 1987 budget in FY 1985. Since that time, it has been determined that the 242-S Evaporator will not operate in the near future and, therefore, the "242-S Evaporator Pond" will not be needed. The second proposed GPP was to construct a "200 East Area Contingency Pond." This pond is necessary in order to allow decommissioning and stabilization of an existing pond which, because of process upsets in the early 1960's, is necessary in order to prevent additional future contamination of the ground water.

Mr. AuCoin. Are funds requested for this work in Fiscal Year 1987?

Adm. Foley. The 200 East contingency pond at Hanford will be constructed with an FY 1987 requested funding of $885K. It is scheduled for completion by January 1987.

WASTE DISPOSAL

Mr. AuCoin. How long does the Department intend to use these new bins--as well as the old one--at Hanford?

Adm. Foley. Disposal of liquid waste with concentrations exceeding drinking water standards will continue at Hanford at a rate commensurate with implementation of the long-range program to reduce low-level waste discharges to the soil. The engineering study of alternatives to this practice is nearing completion. The length of time necessary to complete the implementation of alternative disposal methods is dependent upon the availability of funds, the length of time necessary to construct facilities, and the relative merit of the alternatives as compared with the environmental, health, and safety effects of the current practices. It is anticipated that the long-range program to eliminate discharges above drinking water standards will require a minimum of 10-15 years to complete. At present, the FY 1987 budget supports design work necessary for an $81 million N Reactor project to treat a high-volume, low-level waste stream.

Mr. AuCoin. Why has the Department apparently decided to halt the use of seepage bins at Savannah River by 1988, while extending their use at Hanford?

Adm. Foley. A comprehensive decision to halt the use of seepage basins at Savannah River has not been made. An environmental impact statement is being prepared and will be the basis for decisions on waste management activities for ground water protection at that facility.

Because of ground water contamination, discharges to the M area seepage basins were discontinued in 1985 and will be discontinued to the F and H areas in 1988. Seepage basins are replaced by waste treatment facilities that process effluents for release to the environment in accordance with applicable reguations. Seepage basins continue to service in the C, D, K, P, and L areas of the plant.

Ground water contamination is not a health and safety problem at Hanford. With low annual rainfall and a large distance to the ground water, the Hanford climate provides for an efficient solar process to evaporate water from the liquid waste before it would reach the ground water. Furthermore, no ground water at the Hanford site is used for human consumption. The ground water flowing beneath the site discharges to the Columbia River where levels for hazardous and radioactive material are well below drinking water limits.

Mr. AuCoin. What studies have been conducted independently of the Department to monitor the migration of wastes in ground water around the Hanford facility?

Adm. Foley. The safety of current waste management operations has been independently documented on several occasions, including the report published by the National Academy of Sciences in 1978 entitled "Radioactive Waste at the Hanford Reservation - A Technical Review." In addition, there have been other independent reviews of the program. The U.S. Geologic Survey has collected samples from wells and performed analyses from 1974 to 1985. These data are published annually. The State of Washington has also collected and analyzed ground water samples since 1972. The data is published in an annual report. Samples of river water are collected and analyzed routinely by the states of Oregon and Washington to assess the potential impacts of any ground water contamination on the Columbia River. Their data has also been published. The environmental group Greenpeace participated with the States of Oregon and Washington in a river sampling program in 1985 to validate results reported by the DOE and the Pacific Northwest Laboratory. The results showed that all samples taken were consistent with previously reported data and well within applicable Federal and State standards.

Mr. AuCoin. Can soil be used as a disposal for commercial reactors?

Adm. Foley. Commercial reactors, licensed by the Nuclear Regulatory Commission (NRC), are normally located on small sites dedicated to their use. The Department's production facilities are generally located on much larger sites and generate greater quantities of waste than commercial reactors. In both cases, contaminated effluents are routinely and unavoidably released to the environment.

For commercial reactors, guidelines for releases are established under Chapter 10, Part 50, Appendix I of the Code of Federal Regulations, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet ALARA Criteria for Radioactive Materials in Light-Water Cooled Nuclear Power Reactor Effluents." The utility can also apply to the NRC under Chapter 10, Part 20.302 to obtain approval for onsite disposal in soil, of certain quantities of radioactive waste. There is no regulation prohibiting a utility from disposing of its waste in soil columns onsite. The Department's releases are made consistent with its own orders, which provide comparable levels of safety and environmental protection.

The NRC allows its licenses to operate a variety of facilities for the management of low-level contaminated liquid effluents, including settling ponds, evaporation ponds, and drainage canals. In settling ponds, contaminated effluents are stored to allow the settling of solids suspended in the waste. The slightly contaminated liquids are then discharged to the environment via nearby streams, lakes, or other bodies of water. The solids may then be dredged and are disposed of either onsite or offsite. With an evaporation pond, low-level liquid effluents are allowed to evaporate by solar heating. With drainage canals, low-level liquid effluents percolate through the soil. The NRC evaluates such practices on a case-by-case basis, considering offsite doses to the maximally exposed individual.

Usually, commercial reactors ship their higher activity waste offsite to disposal facilities in Washington State, Nevada, or South Carolina, which are licensed by the respective state under the NRC's Agreement State program. Chapter 10, Part 61, the Federal regulation under which land disposal facilities operate, states in Section 61.56, Waste Characteristics, that this waste must be solidified prior to disposal. Existing commercial disposal facilities do not accept liquid waste for disposal in soil columns.

Mr. AuCoin. Is soil used as medium at Hanford?

Adm. Foley. The soil column has been used for disposal of low-level liquid waste at Hanford since initial operation at the site in the 1940's. The basic nature (alkalinity) of the soil, its ability to chemically bind most radionuclides, the arid climate and long distance to the ground water, and the DOE's continuing commitment to reducing the amount of contaminants disposed of to the soil column have combined to minimize the nature and extent of environmental impacts from this practice. This is a safe and effective means of disposing of such waste at Hanford as documented in ERDA-1538, "Final Environmental Statement - Waste Management Operations," December 1975, and in annual site-wide environmental monitoring reports.

Mr. AuCoin. What is the expected life of single-shell waste containers?

Adm. Foley. There are two factors affecting expected life: the reinforced concrete shell (which provides structural integrity for the tank) and the carbon steel liner (which provides the integrity for liquid storage). Studies and monitoring of the reinforced concrete shell verify that there has been no degradation of the concrete, or the structural integrity of the tank, and that none is expected in the foreseeable future. The life of the carbon steel liners is much less predictable and is the principal reason for programs commenced in the 1960's to remove water from single-shell tanks, leaving behind solids, with storage of the concentrated liquids in new double-shell tanks. The Department is in the last phases, that is Stabilization and Isolation, of liquid removal from these tanks, with completion expected by the mid-1990's. The single-shell tanks are not considered reliable for storage of liquid waste. However, their expected life is virtually unlimited for storage of solids remaining after completion of the Stabilization and Isolation program.

RCRA PROVISIONS

Mr. AuCoin. Why is the Department not subject ot regulation under ECRA for disposal of non-nuclear wastes?

Mr. Salgado. On the basis of certain provisions in RCRA, the Department originally believed that RCRA did not apply to DOE's Atomic Energy Act defense facilities. However, the Department's position since 1984, and today, is that RCRA does apply to these facilities and the Department is subject to regulation under RCRA for the disposal of non-nuclear hazardous wastes.

Mr. AuCoin. Why is the Department not subject to RCRA regulation for disposal of mixed waste?

Mr. Salgado. Similar to the preceding answer, although the Department originally believed that RCRA did not apply to its defense facilities generating mixed wastes, the Department's position changed in 1984. The Department felt that the dual character of mixed wastes warranted dual regulation: regulation under RCRA to guard against its chemical hazard, and regulation under the Atomic Energy Act (AEA) to guard against its radiological hazard. Where the application of a particular RCRA requirement is inconsistent with the AEA, as Section 1006(a) of RCRA anticipates, adaptations may be required. This remains the Department's position today.

WASTE DISPOSAL

Mr. AuCoin. How does the Department plan to store waste that cannot be permanently stored in single-shell containers?

Adm. Foley. In April 1986, the Department issued, for public reveiw and comment, the draft Hanford Defense Waste Environmental Impact Statement, which explores a range of alternatives for permanent disposal of waste that is presently stored in the single-shell tanks, as well as other defense waste stored at Hanford. The alternatives for single-shell tank waste dispoal range from disposal in-place, with engineered barriers, to retrieval of the waste, processing, and disposal in a repository. Environmental impacts for these alternatives are similar. The costs for the retrieval option are approximately 5-10 times higher.

The schedule for completing the NEPA process calls for resolution of comments and completion of the Environmental Impact Statement in the summer of 1987, followed by a Record of Decision that will form the basis for budgetary requests to implement decisions resulting from the NEPA process.

BORON AVAILABILITY

Mr. Watkins. It is my understanding that the isotope Boron-10 of the element Boron, due to its ability to absorb neutrons, is the only element capable of shutting down a nuclear meltdown. It is also my understanding that natural Boric Acid in a concentration of 13 percent can shut down a nuclear reaction, but that the use of Boron-10 rather than natural Boron to make the solution increases effectiveness by a factor of five, permitting the shut down of a nuclear reaction five times as fast when a Boron-10 solution is used as compared to when natural Boric Acid solution is used. Please comment on the accuracy of this information, and provide your perspective of the benefits of using Boron-10 and natural Boric Acid to stop nuclear meltdowns.

Adm. Foley. Boron is only one of a number of naturally occurring elements that, due to their ability to absorb neutrons, are capable of shutting down a nuclear reaction. Certain other characteristics could make it useful for some emergency situations, but it is by no means the only possible choice. The isotope Boron-10 accounts for approximately one-fifth of naturally occurring Boron. The rest has essentially no ability to absorb neutrons. Consequently, in those instances where the volume of material is important to the control of a nuclear reactor, Boron-10 (undiluted by the presence of Boron-11, the other isotope of naturally occurring Boron) is a more effective control material than is natural Boron. However, the circumstances where material volume is important are rare, and natural Boron is generally sufficient for nuclear control.

With regard to use of Boron, in any form, to control a nuclear meltdown, it is important to realize that the effect of the boron is to terminate the chain reaction. Unfortunately, heat continues to be generated in nuclear fuel from the radioactive decay of fission products even after the chain reaction has been stopped. Meltdown, if it happens, occurs because there is insufficient cooling to remove this heat. Consequently, even though Boron is quite effective in terminating the chain reaction, provision of a cooling mechanism is generally the most important factor in stopping a nuclear meltdown.

Mr. Watkins. Is the use of Boron-10 considered to be an essential component of our nation's present nuclear reactor safety system?

Adm. Foley. The use of Boron-10 is not considered to be an essential component of our nation's present nuclear reactor safety systems.

Mr. Watkins. Of other nations' nuclear safety systems?

Adm. Foley. To our knowledge, Boron-10 is not considered to be an essential component of other nations' nuclear safety systems.

Mr. Watkins. What role does both Boron-10 and natural Boric Acid play in national and international safety systems?

Adm. Foley. Natural Boric Acid, as a soluble form of natural Boron, is used in many pressurized water reactors as a supplemental control system during normal operation to minimize movement of the control rods. It is used in the ECCS water supply. Boron-10 is not widely used in safety systems, except as one of the components of natural Boron. Natural Boron is widely used throughout the world in nuclear reactor safety systems.

Mr. Watkins. Is it currently used, or being considered for use, as a nuclear reactor standby liquid control system?

Adm. Foley. Natural Sodium Pentaborate rather than Boric Acid is used as a nuclear reactor standby liquid control system for Boiling Water reactors.

Mr. Watkins. What plans exist for its current or future use?

Adm. Foley. Natural Boron, either in solid form (generally as Boron Carbide) or in solution (as Boric Acid) will likely continue to be used in nuclear reactor control systems for many years to come, largely because it is relatively inexpensive, can be used either as a solid or a liquid, and, in its solid form as Boron Carbide, is relatively unaffected by heat or high temperature. Boron-10, on the other hand, is relatively expensive and does not offer significant advantages over natural Boron. Therefore, future demand for it will be relatively low and dependent upon specialty applications.

Mr. Watkins. What means other than the use of Boron-10 exist for stopping nuclear meltdowns?

Adm. Foley. As discussed earlier, the best method for stopping a nuclear meltdown is the provision of adequate cooling. Apart from that, there are a number of ways that might be considered to stop a nuclear chain reaction before a meltdown occurred. These generally involve the rapid insertion of a control material, such as Boron. Materials such as Indium, Cadmium, Hafnium, Samarium, Gadolinium, Europium, Dysprosium, and Lithium could be used in most production and power reactors. Other reactor types may require control systems that manipulate neutrons that escape from reactor and materials such as

Beryllium, Carbon, and Iron. Once the fuel has started to melt, terminating any continuing chain reaction requires consideration of both the geometry and the material distribution, and will depend greatly upon the specifics of the situation.

Mr. Watkins. What are the relative costs and benefits of each of these nuclear reactor safety systems?

Adm. Foley. The relative costs of the various nuclear reactor safety systems are strongly dependent upon the specific design of the system. The systems that employ control materials such as Boron represent a tiny fraction of the cost of the plant. Qualitative benefits of the various materials are shown in a table that I would like to insert in the record.
(The information follows)

QUALITATIVE BENEFITS OF VARIOUS MATERIALS

	Neutron Absorption (Thermal)	Neutron Absorption (Higher Energy)	High Temperature Stability
Boron	High	Medium	Good
Cadmium	High	Low	Moderate
Dysprosium	Medium	High	Good
Europium	Medium	High	Good
Gadolimium	High	Low	Good
Hafnium	Medium	Medium	Moderate
Indium	Medium	High	Moderate
Luthium	High	Medium	Moderate
Samarium	Medium	High	Good

Mr. Watkins. Is not customized isotopic enrichment of Boron-10 the most cost-effective choice for both nuclear control systems and radioactive waste management?

Adm. Foley. No, enriched Boron is not necessarily the most cost-effective choice. For most applications, natural Boron would be equally effective and less costly.

Mr. Watkins. Does the U.S. government and/or the American nuclear industry currently maintain stockpiles of Boron, Boric Acid, or Boron-10 for use in the event of a nuclear reactor emergency? If so, what amount of each form of Boron is currently in our stockpiles? How has it been determined what amount of Boron-10 should be stockpiled?

Adm. Foley. Neither the U.S. government nor the American nuclear industry maintain stockpiles of Boron, Boric Acid or Boron-10 for use in the event of a nuclear emergency.

Mr. Watkins. Is it not true that it takes 60 days to begin yielding production at a Boron enrichment facility, thus requiring production of Boron-10 for use in nuclear accidents before any such emergency occurs?

Adm. Foley. As is true with any production facility, product output generally is several days delayed from feed material input because of the processing time. For a Boron-10 production facility, the time would also be dependent upon the condition of the plant and whether or not it required refurbishment, startup, etc. In any event, production would indeed lag input.

Mr. Watkins. When does DOE forsee the need for production of additional quantities of Boron-10?

Adm. Foley. There does not appear to be any DOE programmatic need to produce additional quantities of Boron-10 for at least the next several years.

Mr. Watkins. Where are current Boron-10 stockpiles stored?

Adm. Foley. Neither the U.S. Government nor the American nuclear industry maintain stockpiles of Boron-10.

Mr. Watkins. How readily accessible is it to U.S. nuclear reactor sites?

Adm. Foley. Since there are no stockpiles, it is not readily accessible.

Mr. Watkins. What procedures are in place for use of our stockpiled Boron-10 in the event of a nuclear accident in the United States?

Adm. Foley. No such procedures are in place.

Mr. Watkins. In other countries?

Adm. Foley. There are no procedures in place in other countries that I am aware of.

Mr. Watkins. It has been reported that Moscow party leader Boris N. Yeltsin reported that helicopters have dropped sacks of sand, lead, and Boron at the Chernobyl nuclear power plant in the Soviet Union in an effort to cut emissions of radioactivity at the site to a minimum. How has natural Boron and/or Boron-10 been used in connection with the Chernobyl nuclear accident?

Adm. Foley. Details of the Chernobyl accident are not clearly known. However, if Boron and/or Boron-10 were used in connection with the Chernobyl accident, it was probably an attempt to assure there was no circumstance under which a nuclear chain reaction could occur as the soviet's attempted to bring an end to the uncontrolled release of radioactivity.

Mr. Watkins. What was the source of the Boron used by the Soviets?

Adm. Foley. If Boron has been used by the Soviets, its source is unknown at this time.

Mr. Watkins. Does the U.S. make its supply of Boron available to the Soviet Union and the rest of the world for use in shutting down nuclear reactors?

Adm. Foley. Neither the U.S. Government nor the American nuclear industry maintain stockpiles of Boron.

Mr. Watkins. It is my understanding that the Boron-10 facility in Quapaw, Oklahoma, which is owned and operated by Eagle-Picher Industries, Inc., is the only Boron enrichment facility (capable of separating natural Boron into the isotopes Boron-10 and Boron-11) in the Western hemisphere. It is also my understanding that this facility was constructed as a Department of Energy facility, and was mothballed and placed on standby status by DOE on December 1, 1983, causing the lay-off of 300 employees. Please verify this information, and comment on current DOE plans for use of this facility in the future.

Adm. Foley. Applications requiring boron enriched in the isotope Boron-10 are so volume-limited that the significant cost of enriching boron-10 is not justified. The neutron absorption properties of boron-10 are about five times greater than natural occurring boron. At the present time, all applications for the enriched boron-10 have been satisfied. It is true that the facility was built as a Department of Energy facility and was closed in 1983.

Mr. Watkins. When does DOE plan to produce more Boron-10 at the Quapaw facility?

Adm. Foley. There are no plans for reactivation of this facility.

Mr. Watkins. Is use of the Eagle-Picher Boron facility being considered for use by DOE for other applications of the various forms of Boron?

Adm. Foley. Outside of extremely limited uses of boron-10 in radiation detectors and specialized nuclear diagnostic equipment, there are no present uses for boron-10 that have been identified.

Mr. Watkins. What efforts is DOE making to explore other uses of the various forms of Boron?

Adm. Foley. There are no forseen uses for other isotopes of boron. It is possible that there may be a future application for Boron-10, such as control rods for a space-based nuclear reactor. At that time, sources for enriched Boron-10 could be considered.

Mr. Watkins. Has Boron-10 or natural Boron ever been used by the United States for safety reasons at any nuclear reactor facility? If so, when and where was it used?

Adm. Foley. Natural Boron is used in the safety system of the Hanford N Reactor and was used in the safety systems of the earlier Hanford reactors. Natural Boron is used in the safety systems of most Boiling Water Reactors. Sodium Pentaborate is used in the "Standby Liquid Control System" of many Boiling Water Reactors. Natural Boric Acid is used in the Emergency Core Cooling System water supply of Pressurized Water Reactors.

There have been no events in the United States that compare with the Chernobyl event, and so Boron has not been used in the United States for such a purpose. There have been three reactor events that resulted in some fuel melting. These are the SL-1 Military reactor accident in Idaho in the 1960's, the Fermi Power Plant coolant blockage, and the TMI accident. In none of these events was boron used, in addition to existing safety systems, during or immediately following the event. In the case of the SL-1 accident, borated water was used to stabilize the system after the condition of the fuel and reactor vessel were known which was substantially after the event.

Mr. Watkins. What circumstances prompted its use?

Adm. Foley. The use of Boron, as discussed earlier, has been very successful.

Mr. Watkins. How successful was it?

Adm. Foley. As discussed earlier, natural Boron is widely and successfully used throughout the nuclear industry.

MR. BEVILL: CONSIDERING THE LIMITED AMOUNT OF INFORMATION AVAILABLE FROM THE USSR, COULD YOU PROVIDE THE COMMITTEE WITH A GENERAL DESCRIPTION OF THE ACCIDENT IN CHERNOBYL?

CHAIRMAN PALLADINO:

WE DO NOT YET HAVE ANY AUTHORITATIVE DESCRIPTION OF THE CHERNOBYL ACCIDENT. THE SOVIET GOVERNMENT HAS ONLY STATED THAT THE MOST PROBABLE CAUSE OF THE ACCIDENT WAS A CHEMICAL EXPLOSION IN THE REACTOR.

BASED ON THE INFORMATION AVAILABLE TO DATE, WE CAN CONSTRUCT THE FOLLOWING:

AT 1:23 A.M. ON APRIL 26, 1986, AN EXPLOSION OR A SERIES OF EXPLOSIONS OCCURRED IN THE REACTOR BUILDING WHICH HOUSED THE CHERNOBYL NUCLEAR PLANT, UNIT 4. ALL THAT IS KNOWN ABOUT THE PRIOR REACTOR STATUS IS THAT IT HAD BEEN OPERATING AT 7 PERCENT POWER. THE ACTUAL CAUSE AND LOCATIONS OF THE EXPLOSION(S) ARE UNKNOWN BUT THE CONSEQUENCE OF THE EXPLOSION(S) LED TO FAILURE OF THE PRIMARY SYSTEM BOUNDARY, RADIOACTIVE RELEASE AND A SUSTAINED GRAPHITE FIRE. TEMPERATURES PRODUCED WERE SUFFICIENT TO DESTROY THE FISSION PRODUCT RETENTION CAPABILITY OF THE FUEL, AND FROM THE OBSERVED SPECTRUM OF RELEASED FISSION PRODUCTS AND THE COURSE OF LATER EVENTS FUEL SLUMPING AND MELTING CAN BE INFERRED. THE GRAPHITE FIRE HAS BEEN EXTINGUISHED AFTER AN UNKNOWN QUANTITY OF GRAPHITE HAD BURNED AND ACTIONS HAVE BEEN TAKEN TO RETAIN MOLTEN CORE MATERIAL FROM ENTERING THE EARTH. THESE ACTIONS INCLUDED FILLING THE SUPPRESSION POOL CAVITY WITH CONCRETE AND INJECTING NITROGEN INTO THE CORE REGION.

THERE ARE MANY POSSIBLE SCENARIOS LEADING TO THE CHERNOBYL ACCIDENT CONSEQUENCES, BUT IT IS NOT YET POSSIBLE TO DEVELOP WITH ANY CONFIDENCE A MORE SPECIFIC DESCRIPTION OF THE CAUSE AND

SUBSEQUENT PROGRESSION OF THE CHERNOBYL ACCIDENT THAN GIVEN ABOVE. WHILE IT IS IMPORTANT SUCH A DESCRIPTION BE FORMULATED AND LATER FOLLOWED WITH A DETAILED AND MECHANISTIC HISTORY, SUCH AN ATTEMPT AT THIS TIME COULD BE MISLEADING AND EVEN COUNTER PRODUCTIVE TOWARD THE EVENTUAL ESTABLISHMENT OF THE CORRECT DESCRIPTION.

BASED ON THE PAUCITY OF ACCURATE INFORMATION REGARDING THE EXACT SEQUENCE OF EVENTS DURING THE FIRST SEVERAL WEEKS FOLLOWING THE TMI-2 ACCIDENT, IT IS LIKELY THAT THE SOVIET GOVERNMENT DOES NOT YET HAVE A COMPLETE, VERIFIED, SEQUENCE OF EVENTS.

IN OUR STUDY OF THE ACCIDENT WE ARE TAKING AN APPROACH THAT DEALS WITH THE COMPONENTS OF SCENARIOS. THESE COMPONENT GROUPINGS ARE LISTS OF POSSIBLE INITIATORS, PROGRESSION SEQUENCES, CHEMICAL REACTIONS, ACCIDENT INTENSIFIERS, AND TERMINATION MECHANISMS. WE HAVE NOT ATTEMPTED TO SELECT AND ASSEMBLE THESE COMPONENTS INTO ONE OR MORE LIKELY SCENARIOS BECAUSE WE CANNOT BE CONFIDENT THAT ANY OF THESE GROUPS ARE COMPLETE OR THAT WE CAN ACCOUNT FOR SIGNIFICANT HUMAN ERRORS THAT COULD CAUSE OR AGGRAVATE THE ACCIDENT. RATHER, THE COMPONENT GROUPINGS ILLUSTRATE WHAT COULD BE POSSIBLE AND WHAT ARE THE DRIVING FORCES FOR INTENSIFICATION AND TERMINATION.

WE ARE CONTINUING TO MONITOR INFORMATION RELEASES FROM THE SOVIET UNION IN ORDER TO INCREASE OUR UNDERSTANDING OF THE EVENT. HOWEVER, ANY FURTHER DESCRIPTION OF THE EVENT CAUSE AND SEQUENCE AT THIS TIME IS ONLY SPECULATION.

ON MAY 13, 1986, THE NRC'S CHERNOBYL INCIDENT TRACKING TEAM BRIEFED THE COMMISSION ON THE CHERNOBYL INCIDENT. THIS BRIEFING PACKAGE IS ATTACHED AND PROVIDES A COMPREHENSIVE SUMMARY OF OUR KNOWLEDGE OF THE CHERNOBYL SITE, PLANT DESIGN, ACCIDENT SEQUENCE AND STATUS, AND RADIOLOGICAL IMPACT AS OF THAT DATE.

SITE AND PLANT DESCRIPTION

THE ACCIDENT

- EVENT DESCRIPTION SUMMARY AND STATUS

- RADIOLOGICAL DATA

FUTURE PLANS

DESCRIPTION OF THE CHERNOBYL SITE

- THE CHERNOBYL REACTOR SITE IS A 4 UNIT SITE LOCATED ON THE PRIPYAT RIVER APPROXIMATELY 10 MILES NORTHWEST OF CHERNOBYL

- CHERNOBYL IS A SMALL TOWN LOCATED APPROXIMATELY 60 MILES NORTH OF KIEV

- POPULATIONS

    - KIEV - 2½ MILLION

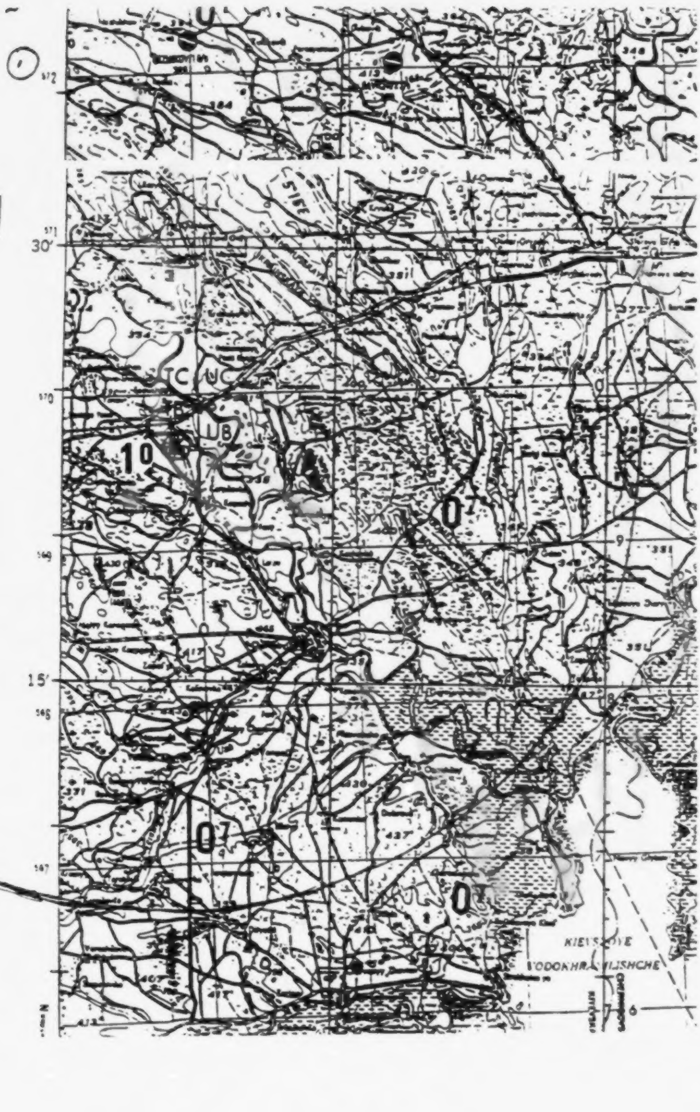
    - PRIPYAT + 3 OTHER NEARBY TOWNS - 49,000
      WITHIN 18 MILE RADIUS - 150,000 TO 180,000

    - WITHIN A 100 MILE RADIUS - 7 MILLION

- TERRAIN APPEARS TO BE ROLLING HILLS

- ALL FOUR UNITS ARE RBMK-1000 CLASS PLANTS. UNIT 1 OPERATIONAL AROUND 1981, UNIT 4 OPERATIONAL IN 1983. TWO MORE RBMK-1000 CLASS UNITS CURRENTLY UNDER CONSTRUCTION AT SITE.

- APPROXIMATELY 1 DOZEN RBMK-1000 CLASS UNITS CURRENTLY IN OPERATION IN SOVIET UNION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30'
TC
UC
UB
15'
KIEVSKOYE
VODOKHRANILISHCHE

CHERNOBYL PLANT CHARACTERISTICS*

- RBMK-1000 PLANT DESIGN IS SIGNIFICANTLY DIFFERENT FROM COMMERCIAL U.S. REACTORS

- THE CHERNOBYL UNITS (RUSSIAN NAME RBMK-1000) ARE DIRECT CYCLE BOILING-WATER PRESSURE-TUBE REACTORS

- EACH UNIT IS RATED AT ABOUT 3200 MWT

- THE REACTOR FUEL IS CONTAINED IN A LARGE NUMBER (~1660) OF INDIVIDUAL ZIRCALOY PRESSURE TUBES IMBEDDED IN A MATRIX OF GRAPHITE BLOCKS.

- EACH PRESSURE TUBE CONTAINS 18 ZIRCALOY CLAD UO_2 FUEL PINS ENRICHED TO ABOUT 1.8% U-235. THE PRESSURE TUBES ARE 8.8 CM (3.5 INCH) IN DIAMETER.

- THE REACTOR IS ABOUT 12M (~40 FT) IN DIAMETER AND 8M (~26FT) HIGH INCLUDING SIDE AND TOP GRAPHITE REFLECTORS

- THE GRAPHITE MATRIX IS ENCLOSED WITHIN AN INERTED ATMOSHPERE

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*ALL INFORMATION OBTAINED FROM AVAILABLE LITERATURE

- SIX OPERATING PUMPS (8 PUMPS TOTAL, TWO PUMPS ARE ON STANDBY) CIRCULATE COOLING WATER TO THE INLET OF EACH INDIVIDUAL PRESSURE TUBE (FUEL ASSEMBLY) THROUGH INDIVIDUAL LINES FOR EACH ASSEMBLY

- STEAM IS PRODUCED WITHIN THE ASSEMBLY AND IS EXTRACTED FROM THE TOP OF EACH PRESSURE TUBE AND COLLECTED IN A HEADER COMMON TO ONE OF FOUR STEAM DRUMS

- CONTROL IS ACCOMPLISHED WITH 211 BORON CARBIDE CONTROL RODS

- THE CONTROL ROD CHANNELS ARE COOLED SEPARATELY

- ON LINE REFUELING (ESSENTIALLY CONTINUOUS)

- EMERGENCY COOLING SYSTEMS ARE DESIGNED FOR MAIN COOLANT PIPE BREAK

- 3 DIESEL GENERATORS FOR 2 UNITS

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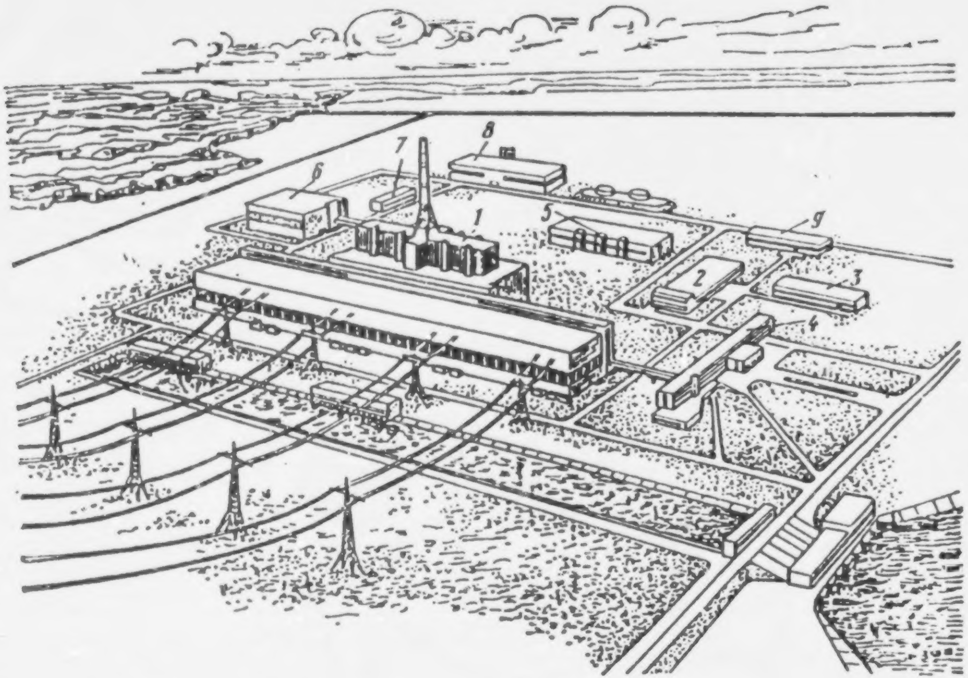

Fig. 3-12. General view of the Smolensk nuclear power plant

1—main building; *2*—auxiliary building and chemical water clean-up building; *3*—chemical storehouse; *4*—administrative and welfare building; *5*—Diesel-generator station; *6*—liquid- and solid-waste storage facility; *7*—nitrogen-oxygen station; *8*—stand-by boiler; *9*—new fuel storage facility

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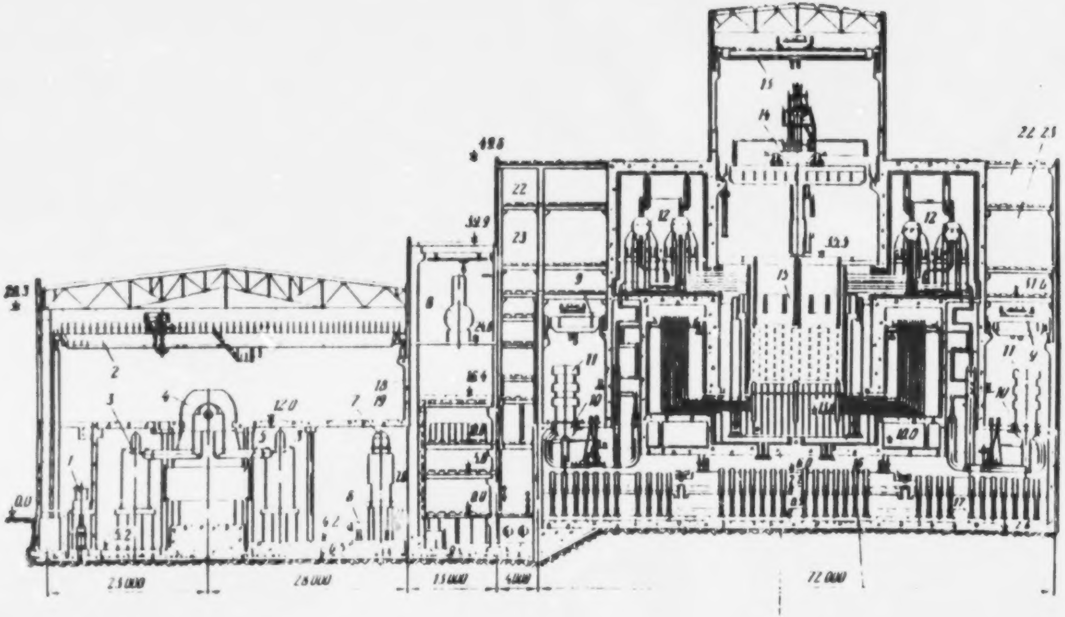
28.3
49.8
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24.0
16.4
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28 000
28 000
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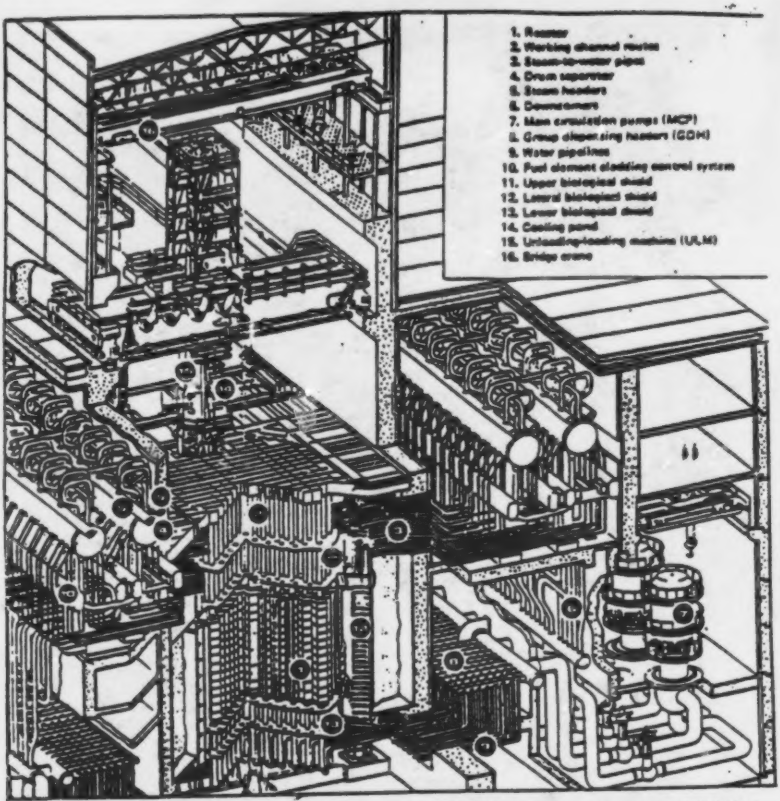
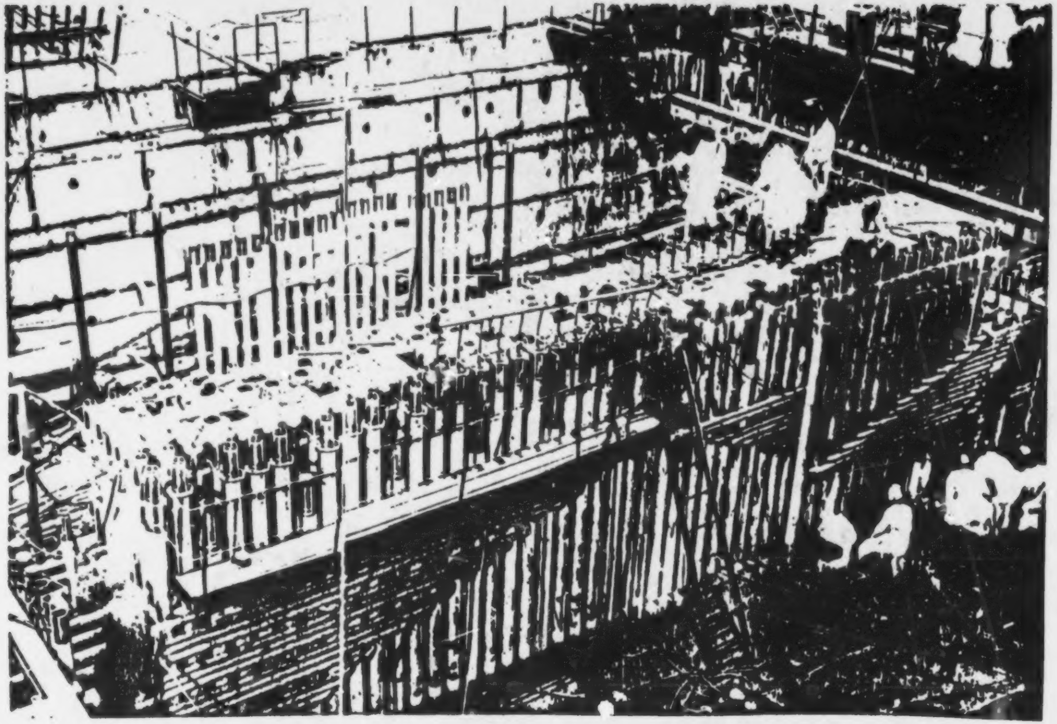
4 000
72 000

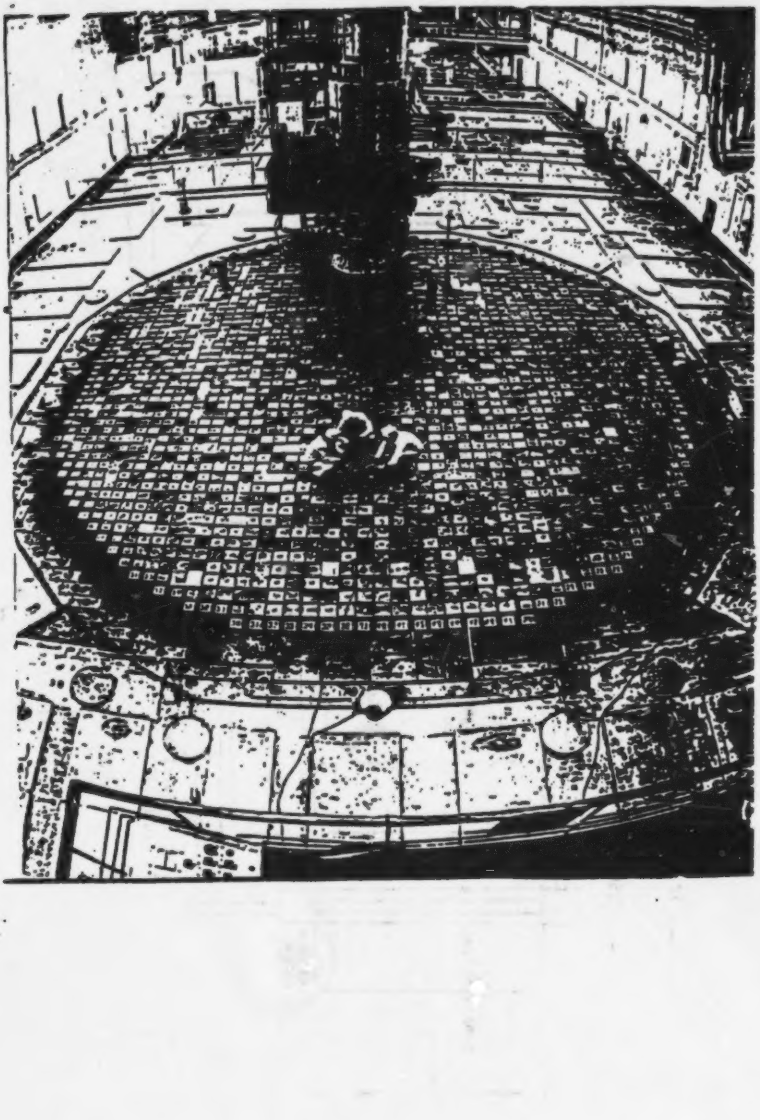
Fig 3.13 Cross sectional view of the main building at Smolensk

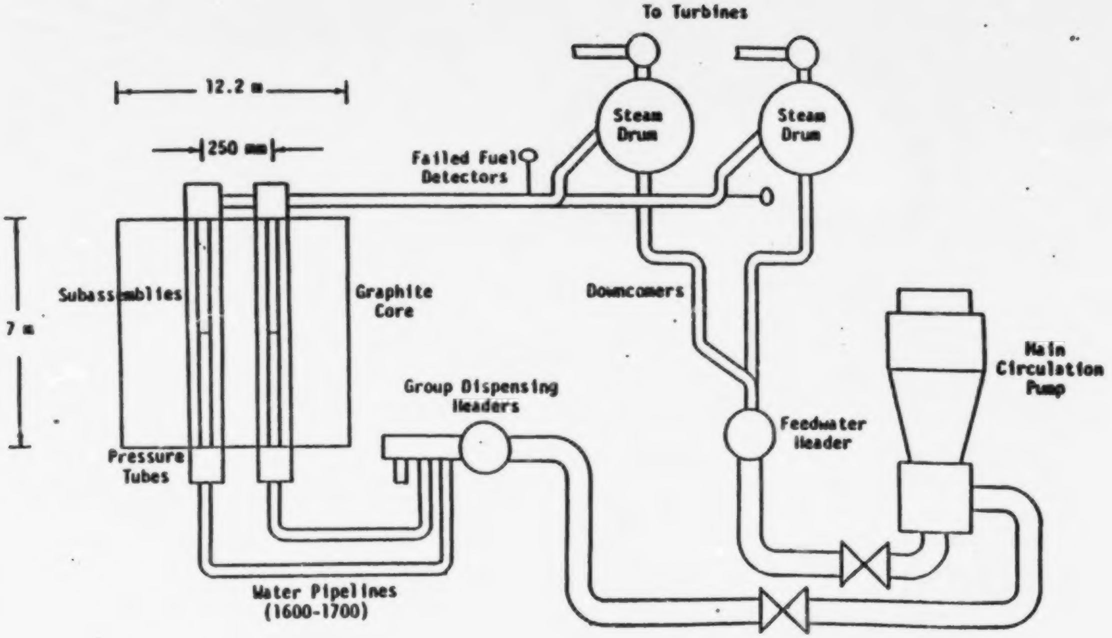
1 first stage condensate pump, 2 125/20 t overhead travelling crane, 3 separator steam superheater, 4 K 500 65/3000 steam turbine, 5 condenser, 6 additional cooler, 7 low pressure heater, 8 deaerator, 9 50/10 t overhead travelling crane, 10 main circulating pump, 11 electric motor of main circulating pump, 12 drum separator, 13 50/10 t remotely controlled overhead travelling crane, 14 refueling mechanism, 15 RBMK 1000 reactor, 16 accident containment valves, 17 bubbler pond, 18 pipe aisle, 19 modular control board, 20 location beneath control board room, 21 house switchgear locations, 22 exhaust ventilation plant locations, 23 plenum ventilation plant locations


1. Reactor
2. Working channel routes
3. Steam-to-water pipes
4. Drum separator
5. Steam headers
6. Downcomers
7. Main circulation pumps (MCP)
8. Group dispersing headers (GDH)
9. Water pipelines
10. Fuel element cladding control system
11. Upper biological shield
12. Lateral biological shield
13. Lower biological shield
14. Cooling pond
15. Unloading-loading machine (ULM)
16. Bridge crane

A sectional view of the RBMK-1000 reactor. With an electrical power of 1000 MW, the reactor's thermal power is 3140 MW; the coolant flow is 37.5 x 10³ t/h and steam capacity 5.4 x 10³ t/h. The reactor inlet water temperature is 270°C and the saturated steam temperature 284°C with a pressure in the separator of 70 kg/cm². The initial fuel enrichment is 1.8%.






To Turbines
12.2 m
250 mm
Steam Drum
Steam Drum
Failed Fuel Detectors
7 m
Subassemblies
Graphite Core
Downcomers
Main Circulation Pump
Group Dispensing Headers
Feedwater Header
Pressure Tubes
Water Pipelines (1600-1700)

Schematic of an RBMK-1000 Generating Unit

DESCRIPTION OF PRIMARY SYSTEM ENCLOSURE

RBMK-1000

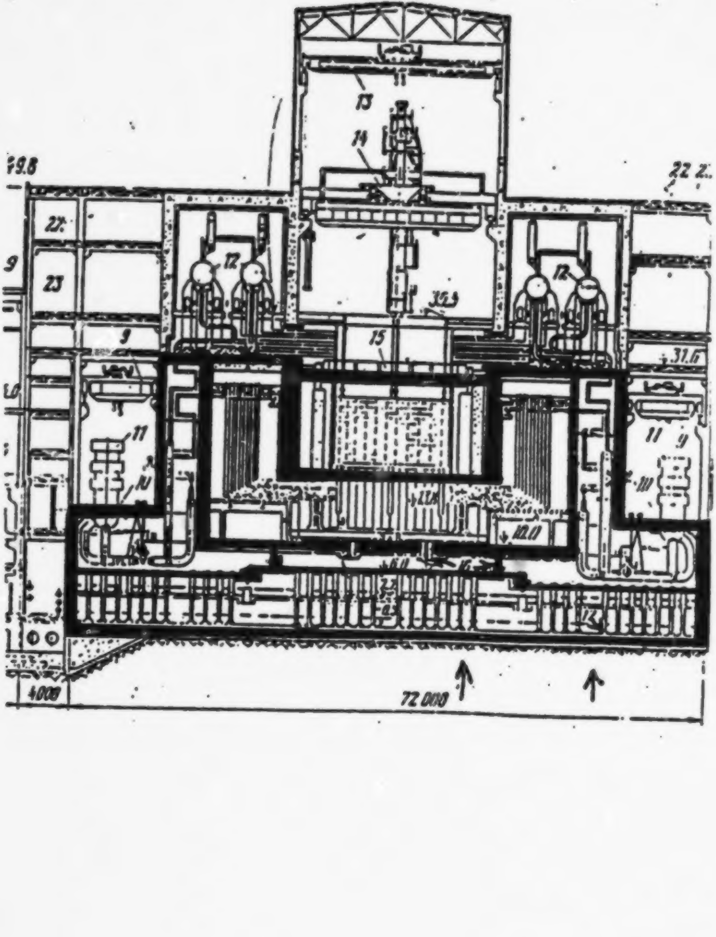
ENCLOSURE PHILSOPHY

DEDUCED FROM LITERATURE

- PRIMARY SYSTEM DOES NOT HAVE HOUSING EQUIVALENT TO A WESTERN STYLE CONTAINMENT
- ENCLOSES REACTOR AND MAJOR PIPING WITHIN MULTIPLE DRY WELLS
- PROVIDES A WETWELL BENEATH THE DRYWELL STRUCTURES
- ENCLOSES REMAINING PRIMARY SYSTEM WITH A CONFINEMENT STRUCTURE
- METAL LINED RECTANGULAR VOLUMES

CURRENT CONCLUSIONS:

- WE DO NOT HAVE ENOUGH INFORMATION TO DRAW ANY DEFINITE CONCLUSIONS REGARDING THE CAPABILITIES OF THE PRIMARY SYSTEM ENCLOSURE
- IT APPEARS THAT THE SOVIET PHILOSOPHY OF CONTAINMENT/ CONFINEMENT IS SIGNIFICANTLY DIFFERENT FROM U.S./EUROPE


13
14
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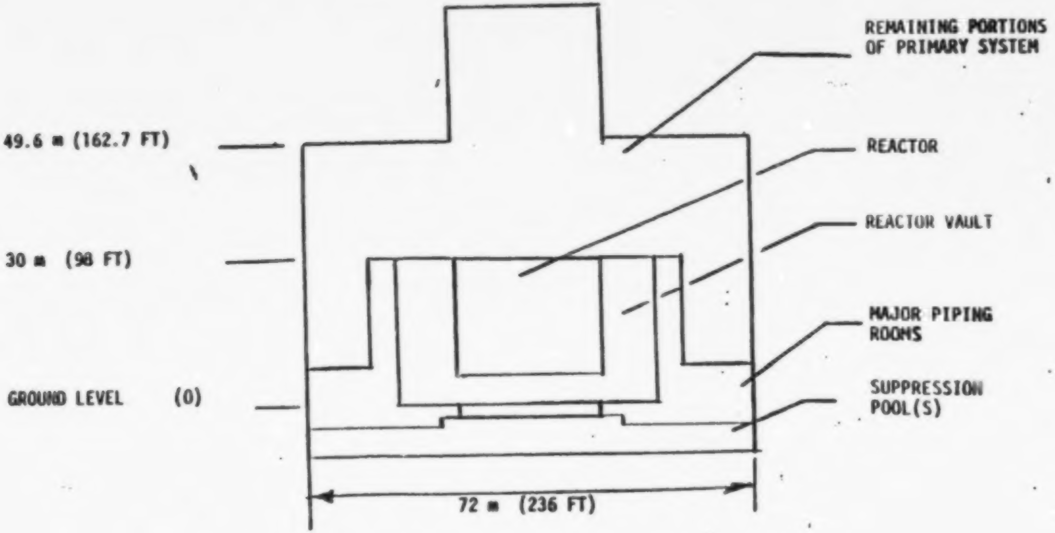
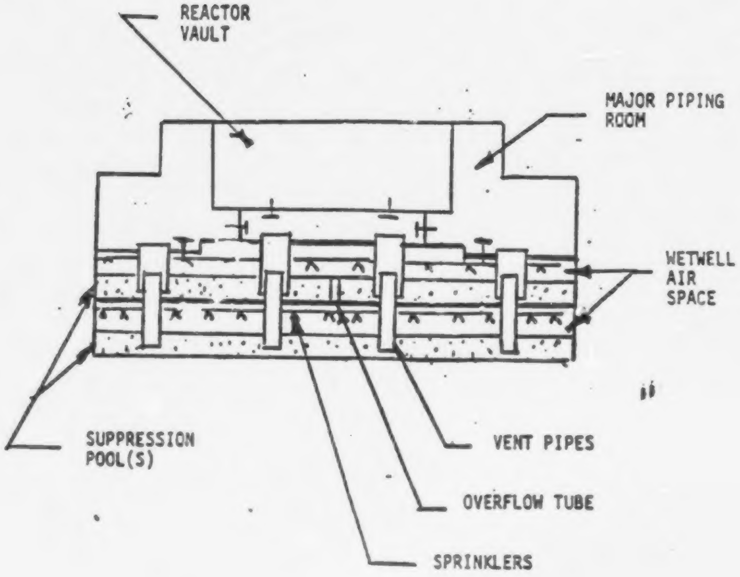
12
15
32.3
31.6
11
10
11
10
4000
72 000

PLANT ARRANGEMENT


49.6 m (162.7 FT)
30 m (98 FT)
GROUND LEVEL (0)
REMAINING PORTIONS OF PRIMARY SYSTEM
REACTOR
REACTOR VAULT
MAJOR PIPING ROOMS
SUPPRESSION POOL(S)
72 m (236 FT)

SUPPRESSION POOL ARRANGEMENT


REACTOR VAULT
MAJOR PIPING ROOM
WETWELL AIR SPACE
SUPPRESSION POOL(S)
VENT PIPES
OVERFLOW TUBE
SPRINKLERS

EVENT DESCRIPTION SUMMARY AND STATUS

- ON APRIL 26, 1986, AT APPROXIMATELY 1:23 AM, EXPLOSION(S) OCCURRED AT THE CHERNOBYL NUCLEAR POWER PLANT IN U.S.S.R.
- REACTOR HAD BEEN PREVIOUSLY OPERATING AT 7% POWER
- PRIMARY SYSTEM BOUNDARY FAILURE PROBABLY CAUSED BY OVERPRESSURIZATION DUE TO SUDDEN POWER INCREASE
- SUBSEQUENT HYDROGEN EXPLOSION(S) AND GRAPHITE FIRE
- DECAY HEAT PLUS HEAT FROM GRAPHITE FIRE PRODUCED TEMPERATURES SUFFICIENT TO MELT FUEL
- GRAPHITE FIRE HAS BURNED DOWN, SOME OR ALL OF MOLTEN CORE MATERIAL IS PROBABLY ON REACTOR CAVITY FLOOR
- CONCERN FOR MOLTEN CORE PENETRATION THRU SUPPRESSION POOL BASEMATS
- SOVIETS HAVE STATED THAT REACTOR IS STABILIZED AND ARE REPORTED FILLING SUPPRESSION POOL CAVITY WITH CONCRETE (WITH INTERNAL COOLING) AND INJECTING NITROGEN INTO CORE REGION
- THERE ARE MANY POSSIBLE SCENARIOS FOR CHERNOBYL THAT CAN LEAD TO THE DAMAGED CORE (BUT STILL DO NOT KNOW WHAT THE ACTUAL SEQUENCE OF EVENTS WAS)

SOURCES OF ENVIRONMENTAL DATA

ELEVATED LEVELS OF RADIOACTIVITY DETECTED IN MORE THAN 10 COUNTRIES

SWEDEN AND FINLAND PROVIDED ACTIVITY MEASUREMENTS AS A

FUNCTION OF TIME BEGINNING SHORTLY AFTER THE ACCIDENT

SOURCE TERM

- SOURCE TERM ESTIMATED BY EXTRAPOLATION OF ESTIMATED DOSE FROM CONCENTRATION MEASUREMENTS IN FINLAND AND SWEDEN BACK TO CHERNOBYL SITE

- ENERGETIC ATMOSPHERIC RELEASE

- AVAILABLE DATA INDICATE PROLONGED RELEASE

EXTRAPOLATION FACTORS

(BACK TO CHERNOBYL SITE)

ACCOUNT FOR:

- DISPERSAL IN AIR
- RADIOACTIVE DECAY
- CLOUD DEPLETION

VALUES FOR:

- EXPOSURE TO NOBLE GASES (1 E+5)
- INHALATION OF IODINES (1 E+6)

ESTIMATED DOSES

	LOCATION	DOSE,* WHOLE BODY	REM THYROID
BASED ON MEASUREMENTS	STOCKHOLM (750 MI)	0.002	0.01
BASED ON MEASUREMENTS	HELSINKI (700 MI)	0.005	NO DATA
BASED ON MODELS	CHERNOBYL (1 MI)	> 100	4000

*BASED ON 1-DAY EXPOSURE IN U.S.S.R.; 2-DAYS FOR SWEDEN AND FINLAND

FUTURE PLANS

- PREPARE REPORT
- ISSUES
    - ° NEED FOR EARLY WARNING SYSTEM (IAEA)
    - ° NEED FOR SYSTEM FOR RAPID INTERNATIONAL DATA COLLECTION AND EXCHANGE (IAEA)
    - ° SEVERE ACCIDENT LESSONS
        - ACCIDENT RELATED ISSUES
        - SOURCE TERM TECHNOLOGY
- NO PROGRAMMATIC CHANGES BASED ON OUR PRESENT KNOWLEDGE AND UNDERSTANDING OF WHAT HAPPENED

MR. BEVILL: CAN YOU DISTINGUISH BETWEEN WHAT IS "KNOWN" AND WHAT IS "ASSUMED" (FACT VS. FICTION)?

CHAIRMAN PALLADINO:

WE DO NOT YET KNOW THE EXACT SEQUENCE OF EVENTS WHICH LED TO THE CHEMICAL EXPLOSION AND GRAPHITE FIRE. SINCE THE ACCIDENT BEGAN, VARIOUS SOURCES HAVE REPORTED CAUSES OF THE ACCIDENT. WE INTERPRET THESE TO BE SPECULATION. WE DO NOT BELIEVE THE ACTUAL SEQUENCE OF EVENTS WILL BE KNOWN UNTIL THE SOVIET GOVERNMENT MAKES THE RESULTS OF ITS INVESTIGATIONS AVAILABLE.

OUR ASSESSMENTS TO DATE SHOW THAT A SPECTRUM OF DIFFERENT INITIATING EVENTS COULD HAVE ULTIMATELY RESULTED IN THE OBSERVED CONSEQUENCES.

MR. BEVILL: I UNDERSTAND THAT THE DESIGN OF THE RUSSIAN REACTOR IS UNIQUE. WOULD YOU COMPARE ITS DESIGN TO THE TYPICAL U.S. CIVILIAN NUCLEAR PLANT?

CHAIRMAN PALLADINO:

THERE ARE TWO MAJOR DESIGN FEATURES OF THE RUSSIAN RBMK-1000 CLASS OF REACTORS THAT ARE UNIQUE COMPARED TO U.S. CIVILIAN LIGHT WATER NUCLEAR POWER REACTORS. FIRST, THE RBMK-1000 USES GRAPHITE TO MODERATE (SLOW DOWN) THE NEUTRONS. WATER IS USED PRINCIPALLY TO PROVIDE COOLING AND, OF COURSE, STEAM FOR THE TURBINE. IN COMMERCIAL U.S. LIGHT WATER REACTORS, WE RELY ON THE WATER BOTH TO MODERATE THE REACTORS AND FOR COOLING AND STEAM PRODUCTION. IN U. S. BOILING WATER REACTORS (BWRS) THIS IS DONE THROUGH A DIRECT-CYCLE BOILING PROCESS (WATER BOILS WITHIN THE FUEL CHANNEL) SIMILAR TO THE RBMK-1000 REACTOR. IN THE RBMK-1000 REACTORS COOLANT FLOW AND STEAM PRODUCTION TAKES PLACE WITHIN MANY (ABOUT 1660) INDIVIDUAL PRESSURE TUBES (AT ABOUT 1000 PSI) WHICH ARE IMBEDDED WITHIN THE GRAPHITE MATRIX. IN BWRS WITHIN THE U.S., COOLANT FLOWS THROUGH A SMALLER NUMBER OF CHANNELS (ABOUT 800 FOR A COMPARABLE POWER LEVEL) ALL OF WHICH ARE CONTAINED WITHIN A PRESSURE VESSEL AT ABOUT 1000 PSI. IN U.S. PRESSURIZED WATER REACTORS (PWRS) COOLANT FLOWS THROUGH OPEN CHANNELS ALSO CONTAINED WITHIN A PRESSURE VESSEL BUT AT HIGHER PRESSURE (ABOUT 2200 PSI) AND STEAM IS PRODUCED IN AN INDIRECT-CYCLE BY BOILING IN SEPARATE ISOLATED FLOW LOOPS WITH ASSOCIATED STEAM GENERATORS.

THE SECOND AND MORE IMPORTANT SIGNIFICANT UNIQUE FEATURE INVOLVES THE REACTOR ENCLOSURE, OR "CONTAINMENT" DESIGN. LITTLE "HARD" ENGINEERING DESIGN DATA IS AVAILABLE BUT, BASED ON INFORMATION OBTAINED FROM PROFESSIONAL JOURNALS, IT APPEARS THAT THE SOVIET CONCEPT OF CONTAINMENT IS SIGNIFICANTLY DIFFERENT FROM THE U.S. CONCEPT. HOWEVER, THERE ARE SEVERAL OBSERVATIONS THAT CAN BE MADE. THE DESIGN DOES NOT ENCLOSE THE ENTIRE PRIMARY SYSTEM AND DOES

NOT APPEAR TO HAVE ANY SIGNIFICANT PRESSURE RETENTION CAPABILITY IN THE EVENT OF A LARGE STEAM RELEASE TO THE REACTOR BUILDING. IT ALSO DOES NOT APPEAR TO HAVE ANY SUBSTANTIAL CAPABILITY TO RETAIN FISSION PRODUCT RELEASES TO THE ENVIRONMENT DURING A SEVERE CORE DAMAGE EVENT.

AS MORE DESIGN INFORMATION BECOMES AVAILABLE, THERE ARE NUMEROUS OTHER DIFFERENCES THAT UNDOUBTEDLY WILL BE CONSIDERED BY US TO BE UNIQUE.

MR. BEVILL: HAS THE ACCIDENT IN THE SOVIET UNION PROVIDED THE U.S. WITH INFORMATION THAT SHOULD MAKE THIS COUNTRY FEEL ITS REACTORS ARE UNSAFE?

CHAIRMAN PALLADINO:

NO. WHILE WE DO NOT YET HAVE ANY DEFINITIVE INFORMATION AS TO THE CAUSE OF THE ACCIDENT, WE HAVE CONCLUDED THAT THE CONSEQUENCES ARE UNIQUE TO THE SOVIET DESIGN AND NOT APPLICABLE TO U.S. LWR DESIGNS.

SPECIFICALLY, U.S. LIGHT WATER REACTORS DO NOT CONTAIN GRAPHITE AND HENCE A GRAPHITE FIRE IS NOT POSSIBLE. U.S. REACTORS ARE ALSO SURROUNDED BY CONTAINMENT BUILDINGS THAT ARE CONTROLLED LEAKAGE STRUCTURES, CAN WITHSTAND INTERNAL PRESSURE AND HAVE SPECIFIC FEATURES DESIGNED TO REDUCE INTERNAL PRESSURES THAT COULD BE GENERATED DURING CERTAIN ACCIDENTS.

WE HAVE ALSO EXAMINED THE APPLICABILITY OF THE SOVIET ACCIDENT TO THE FORT ST. VRAIN REACTOR, WHICH IS THE ONLY GAS COOLED, GRAPHITE MODERATED COMMERCIAL REACTOR IN THE U.S. WE HAVE CONCLUDED THAT THE FORT ST. VRAIN REACTOR HAS SIGNIFICANT DESIGN DIFFERENCES FROM THE SOVIET REACTOR AT CHERNOBYL.

SPECIFICALLY, FORT ST. VRAIN USES A CERAMIC COATED PARTICLE FUEL AND HELIUM GAS AS THE COOLANT. UNLIKE THE CHERNOBYL DESIGN WHICH USES WATER AS THE COOLANT, THE HELIUM USED IN FORT ST. VRAIN CANNOT REACT WITH THE GRAPHITE TO PRODUCE COMBUSTIBLE GASES. FINALLY, THE FORT ST. VRAIN REACTOR IS ENCLOSED IN A PRESTRESSED CONCRETE, STEEL-LINED REACTOR VESSEL WHICH SERVES AS THE CONTAINMENT FUNCTION. THE ONLY WAY TO INTRODUCE SIGNIFICANT QUANTITIES OF AIR INTO THE REACTOR REQUIRES A CONCURRENT FAILURE OF FOUR OF THE REACTOR CLOSURES (TWO EACH, AT THE TOP AND THE BOTTOM OF THE REACTOR VESSEL). THE PROBABILITY OF SUCH AN OCCURRENCE IS CONSIDERED TO BE UNLIKELY.

ONCE THE FACTS REGARDING THE CAUSE OF THE ACCIDENT ARE IN, THE NRC WILL HAVE TO CAREFULLY EVALUATE THEM TO DETERMINE THEIR APPLICABILITY TO U.S. DESIGN REACTORS. FROM THIS EVALUATION AND DETERMINATION WE WILL DECIDE IF AND HOW U.S. REGULATORY REQUIREMENTS SHOULD BE CHANGED.

COMMISSIONER ASSELSTINE ADDS THE FOLLOWING:

I DISAGREE WITH THE COMMISSION'S ASSERTIONS THAT THE OFF-SITE CONSEQUENCES OF THE CHERNOBYL ACCIDENT ARE UNIQUE TO THE SOVIET DESIGN AND NOT APPLICABLE TO U.S. LWR DESIGNS. THE NRC STAFF HAS STATED THAT THE ESTIMATED RADIOLOGICAL RELEASES FROM THE CHERNOBYL PLANT MAY BE EQUAL TO OR LESS THAN ONE OR MORE OF THE FISSION PRODUCT RELEASE CATEGORIES CONTAINED IN THE REACTOR SAFETY STUDY (WASH-1400). MUCH FURTHER RESEARCH AND ANALYSES OF POSSIBLE FISSION PRODUCT RELEASES FOR U.S. PLANTS HAS BEEN ACCOMPLISHED SINCE THAT STUDY WAS COMPLETED IN 1975. HOWEVER, THE NRC STAFF WRITES IN RESPONSE TO QUESTION 13: "TO DATE, INFORMATION (ON POSSIBLE RADIOACTIVE RELEASES FROM CORE MELTDOWNS OF U.S. REACTORS) SUGGESTS THAT SOME CHANGES IN SPECIFIC RADIONUCLIDE GROUP RELEASES TO THE ATMOSPHERE ARE JUSTIFIED; HOWEVER, THE OVERALL CONSEQUENCES ARE NOT SIGNIFICANTLY DIFFERENT FROM THOSE USING SOURCE TERMS CONTAINED IN THE REACTOR SAFETY STUDY." IN EFFECT WE HAVE THE COMMISSION SAYING WE DON'T KNOW MUCH ABOUT THE SPECIFIC DESIGN STANDARDS OF THE CHERNOBYL REACTOR OR THE ACTUAL RELEASES THAT OCCURRED OR ACTUAL OFF-SITE CONSEQUENCES, BUT WE DO KNOW OUR CONTAINMENTS ARE BETTER AND THE CONSEQUENCES OF THE CHERNOBYL ACCIDENT CANNOT HAPPEN HERE AS A RESULT OF A CORE MELTDOWN IN A U.S. REACTOR. IF THE COMMISSION HAS SOUND SCIENTIFICALLY DEFENSIBLE INFORMATION TO SUPPORT ITS ASSERTION THAT THE CONSEQUENCES OF THAT ACCIDENT ARE UNIQUE TO SOVIET REACTORS, THEN THE COMMISSION HAS A DUTY TO PRODUCE IT.

MR. BEVILL: We do not want to underestimate the seriousness of the accident in the USSR, but could you describe, in some detail, what is factually known about the exposure to irradiation of people outside the Soviet Union and please describe the increase in irradiation exposure in non-Soviet Union countries. Compare the potential increase in terms of a typical chest X-ray or a cross country flight.

CHAIRMAN PALLADINO:

Elevated levels of radioactivity have been detected in air, water, soil and food in many countries all over the world. As of May 7, 1986, the NRC has received reports of elevated levels of radioactivity in the following countries: Austria, Belgium, Denmark, Finland, Italy, Japan, Korea, Poland, Sweden, Switzerland, West Germany, and Yugoslavia. There have been no site-specific quantitative data provided by the Soviet Union. Since the previously released radioactive material is continuing to spread, it is expected that additional data will be received from other countries. The radionuclides that have been detected include: Iodine-131, Cesium-137, Cesium-134, Tellurium-132, Ruthenium-103, Molybdenum-99, Neptunium-239, Zirconium-95, and Niobium-95.

Based upon the data received from these sources, the NRC staff has estimated doses to individuals located in several European countries (see Table 1). Exposure to radiation is measured in units of millirem. Doses to the whole body of individuals in European countries (outside the Soviet Union) are estimated to range from less than 1 millirem to greater than 10 millirem. In addition to whole body dose from external irradiation, radioactive iodines concentrate in the thyroid. Doses to the thyroid of individuals from inhalation of air are estimated to range from about 10 millirem to about 500 millirem. The dose decreases as distance from the Chernobyl reactor increases. Exposure from

OTHER PATHWAYS SUCH AS RADIONUCLIDES DEPOSITED ON THE GROUND, AND INGESTION OF FOOD WOULD RESULT IN ADDITIONAL DOSES.

For comparative purposes, the dose from a typical chest X-ray is about 10 to 20 millirem. The dose from a typical cross-country flight would be about 1 millirem.

TABLE 1 Estimated Doses in Europe

Location	Estimated Doses,* Millirem Whole Body	 Thyroid
Belgium	0.2	25
Finland	1 to 10	500
Netherlands	5	**
Poland	10	200
Sweden	2-3	10-500
Switzerland	1-4	**

* Based on exposure for 2 days. The dose to the thyroid is due to inhalation of radioiodines.

** Information not available.

MR. BEVILL: Please describe the exposure limits in this country and compare those to other countries?

CHAIRMAN PALLADINO:

The public health and safety limits for the protection of the general public from exposure to ionizing radiation in the United States and in European countries are based on the recommendations of the International Commission on Radiological Protection (ICRP). In the United States, the safety limits are based primarily on the recommendations of the ICRP promulgated in the year 1959, while many European countries have adopted the ICRP's 1977 recommendations. Although there are some differences between the methodology used in the 1959 and 1977 recommendations, the level of safety provided by either set of recommendations is comparable. A brief description of the dose limits for normal operations and under accident conditions in the United States follows.

For normal operations (i.e., not an accident), the limits for exposure of the general public in the United States to radioactive material are 500 millirem per year to the total body. There are also exposure limits for individual body organs. For example, the dose to the thyroid of an individuals is limited to 1500 millirem per year.

With regard to accidents, the Food and Drug Administration has published recommendations for State and Local agencies regarding accidental radioactive contamination of human food and animal feeds. These recommendations are in the form of preventive and emergency protective action guides (PAGs). The preventive PAGs are dose levels at which protective actions having minimal impact should be taken. Emergency PAGs are dose levels at which more costly protective actions (e.g., isolate food containing radioactivity) should be taken. The preventive PAGs are 1500 millirem to the thyroid, or 500 millirem to the whole body, bone marrow, OR ANY OTHER ORGAN. THE EMERGENCY PAGS ARE TEN TIMES THE VALUES FOR THE PREVENTIVE PAGS. DURING THE COURSE OF THE ACCIDENT AT CHERNOBYL, THE NRC STAFF HAS BRIEFLY REVIEWED THE PROTECTIVE ACTION RECOMMENDATIONS FOR SEVERAL EUROPEAN COUNTRIES. THE RECOMMENDATIONS IN THE EUROPEAN COUNTRIES ARE COMPARABLE TO THOSE IN THE UNITED STATES (I.E., WITHIN A FACTOR OF ABOUT TWO IN MOST CASES).

REP. MYERS: DO WE KNOW FOR CERTAIN WHEN THE ACCIDENT OCCURRED? WHAT IS THE MOST CURRENT INFORMATION ON ITS CONDITION NOW?

CHAIRMAN PALLADINO:

ON MAY 6, 1986, THE ASSOCIATED PRESS REPORTED THAT THE SOVIET GOVERNMENT STATED THAT ON APRIL 26, 1986 AT 1:23 A.M. THE INITIAL CHEMICAL EXPLOSION OCCURRED AS THE PLANT WAS GOING INTO A PLANNED SHUTDOWN. BASED ON RADIOLOGICAL MEASUREMENTS REGARDING THE RELATIVE RATIOS OF IODINE-131 AND IODINE-133, THE NRC STAFF HAS CALCULATED THAT THE PLANT WAS SHUT DOWN AT APPROXIMATELY MIDNIGHT, JUNE 25TH. THIS IS CONSISTENT WITH THE SOVIETS' STATEMENT AND WE ARE NOT AWARE OF ANY OTHER INFORMATION WHICH CONFLICTS WITH THIS REPORT.

BASED ON THE MOST CURRENT INFORMATION AVAILABLE, THE FOLLOWING IS OUR UNDERSTANDING OF THE STATUS OF PLANT AND SITE:

- THE GRAPHITE FIRE HAS BEEN EXTINGUISHED.
- UNIT 4 REACTOR BUILDING IS DESTROYED.
- ACTIONS ARE IN PROCESS TO ATTEMPT TO PREVENT MOLTEN CORE MATERIAL FROM ENTERING THE EARTH.
- RADIOLOGICAL RELEASES CONTINUE FROM THE PLANT, AT LEVELS MUCH REDUCED FROM INITIAL ESTIMATED RELEASE RATES. THE CORE IS MOST LIKELY STILL SMOLDERING.
- EVACUATION WAS OUT TO 18 MILES.
- LIVESTOCK WERE SLAUGHTERED OUT TO 12 MILES.

- 49,000 PEOPLE WERE EVACUATED APPROXIMATELY 60 MILES TO THE SOUTH.
- UNITS 1, 2 AND 3 ARE REPORTED MANNED AND BEING COOLED.

REP. MYERS: Is DOE, NRC, or the State Department advising Americans to avoid travel to any countries that may be affected by this event?

CHAIRMAN PALLADINO:

NRC has issued no advisories regarding travel in the areas affected by this event. The Department of State in conjunction with the Department of Health and Human Services would be the agency responsible for any such statements. On May 3, the Department of State did advise women of child-bearing age and children to avoid travel to Poland until the situation resulting from possible effects of the accident was clarified. The Department of State should be consulted directly for further information on the status of this and other travel advisories.

REP. MYERS: IN LIGHT OF THE INTERNATIONAL OVERTONES OF THIS EVENT, IS THERE ANY SIGNIFICANT DISCUSSION AMONG GOVERNMENTS ABOUT INTERNATIONAL INSPECTION OR EVEN LICENSING OF NUCLEAR PLANTS?

CHAIRMAN PALLADINO:
THERE HAS BEEN NONE TO DATE. NUCLEAR SAFETY EXPERTS OF WESTERN EUROPE, CANADA, JAPAN AND THE UNITED STATES MET IN PARIS ON MAY 9, 1986 TO DISCUSS POSSIBLE IMPLICATIONS OF THE SOVIET ACCIDENT, BUT DID NOT ADDRESS INTERNATIONAL INSPECTION OR LICENSING CONCEPTS. AT THE SPECIAL SESSION OF THE IAEA BOARD OF GOVERNORS CONVENED IN VIENNA MAY 21, IAEA DIRECTOR GENERAL BLIX SUGGESTED SEVERAL WAYS IN WHICH IAEA SAFETY PROGRAMS COULD BE STRENGTHENED, INCLUDING POSSIBLE MECHANISMS FOR EXPANDING THE NUMBER OF ON-SITE REACTOR SAFETY ADVISORY TEAM VISITS TO MEMBER COUNTRIES, BUT HE DID NOT SUGGEST A SYSTEM OF SAFETY INSPECTIONS IN ANY WAY PARALLEL TO THOSE CONDUCTED IN THE SAFEGUARDS AREA. THE SUMMARY STATEMENT ISSUED BY THE CHAIRMAN OF THE BOARD OF GOVERNORS CALLED FOR IAEA MEETINGS TO BE HELD TO REVIEW THE ACCIDENT, TO PREPARE BINDING AGREEMENTS ON PROMPT NOTIFICATION AND EMERGENCY ASSISTANCE IN THE EVENT OF SERIOUS ACCIDENTS, AND TO FORMULATE WAYS OF ENHANCING NUCLEAR SAFETY. IT DID NOT DIRECTLY ADDRESS INTERNATIONAL SAFETY INSPECTION OR LICENSING.

REP. BOGGS: WHAT TYPE OF EMERGENCY RESPONSE PLANS ARE THERE FOR THE POPULATION AROUND THE HANFORD AND SAVANNAH RIVER REACTOR SITES? HOW DO THEY COMPARE WITH OFFSITE EMERGENCY RESPONSE PLANS REQUIRED AT COMMERCIAL REACTOR SITES?

CHAIRMAN PALLADINO:

THE FOLLOWING MATERIAL REGARDING DOE FACILITIES WAS PROVIDED BY DOE IN RESPONSE TO NRC'S REQUEST:

> BOTH THE HANFORD AND SAVANNAH RIVER SITES OF THE DEPARTMENT OF ENERGY (DOE) MAINTAIN COMPREHENSIVE SETS OF EMERGENCY PREPAREDNESS PLANS AND PROCEDURES TO SUPPORT ONSITE AND OFFSITE EMERGENCY RESPONSE ACTIONS IN THE EVENT OF AN ACCIDENT. THESE PLANS AND PROCEDURES ARE TOTALLY INTEGRATED WITH THOSE OF OFFSITE AUTHORITIES. THEY REFLECT CONSERVATIVE ESTIMATES OF THE CONSEQUENCES OF THE MOST SEVERE CREDIBLE ACCIDENT AT EACH NUCLEAR FACILITY ON THESE SITES. A MAJOR EXERCISE IS CONDUCTED ANNUALLY AT BOTH HANFORD AND SAVANNAH RIVER, FREQUENT DRILLS ARE CONDUCTED AT EACH FACILITY, AND ALL PERSONNEL ARE REQUIRED TO PARTICPATE IN THE EMERGENCY PREPAREDNESS TRAINING PROGRAMS. IN ADDITION, EMERGENCY PREPAREDNESS IS INCLUDED AS A TOPIC IN THE EXTENSIVE DOE SITE AND FACILITY SAFETY APPRAISAL PROGRAMS.
>
> THE EMERGENCY PREPAREDNESS PROGRAMS AT DOE SITES, PARTICULARLY AT BOTH SAVANNAH RIVER AND HANFORD, ARE COMPARABLE TO THOSE OF COMMERCIAL REACTOR SITES. ALMOST ALL OF THE EMER-

GENCY PLANNING ZONES (EPZ) FOR THE FACILITIES AT THESE SITES ARE WITHIN THE SITE BOUNDARY; THUS, THE DOE FIELD OFFICE MANAGERS, WHO REPORT DIRECTLY TO THE SECRETARY OF ENERGY, ARE IN CONTROL OF ALL PERSONS WITHIN THE EPZs. THEY ALSO MAINTAIN CLOSE RELATIONSHIPS WITH STATE AND LOCAL OFFICIALS, AND WITH REGIONAL NUCLEAR REGULATORY COMMISSION AND FEDERAL EMERGENCY MANAGEMENT AGENCY OFFICES, TO ASSURE THE PRECAUTIONARY OFFSITE ACTIONS WILL BE EFFECTIVE IF THEY ARE NEEDED IN THE EVENT OF AN ACCIDENT.

FOR COMMERCIAL NUCLEAR PLANTS LICENSED BY NRC, REGULATIONS OF THE NRC AND FEMA REQUIRE EMERGENCY PREPAREDNESS PROGRAMS WITH THE FOLLOWING PRINCIPAL FEATURES:

1. DETAILED PLANS EXIST FOR PROTECTIVE MEASURES FOR THE PUBLIC FOR A PLUME EXPOSURE EMERGENCY PLANNING ZONE (EPZ) OF ABOUT 10 MILES IN RADIUS AND AN INGESTION PATHWAY EPZ OF 50 MILES IN RADIUS. WITHIN THE PLUME EXPOSURE EPZ, SHELTERING AND EVACUATION MUST BE PLANNED FOR. WITHIN ABOUT 50 MILES, THE CONTROL OF MILK PRODUCTS AND OTHER FOOD STOCKS MUST BE PLANNED FOR.

2. SPECIFICATION OF GRADED EMERGENCY CLASSES ARE REQUIRED. THERE ARE FOUR DIFFERENT CLASSES RANGING FROM NOTIFICATION OF AN UNUSUAL EVENT UP TO A GENERAL EMERGENCY. FOR ALL OF THESE CLASSES, STATE AND LOCAL GOVERNMENTS MUST BE NOTIFIED BY THE

utility within about 15 minutes. For the most severe class the public also must be notified and kept updated during the course of any accident.

3. State and local government authorities are required to have the capability to notify the public within 15 minutes after notification by the licensee.

4. Each nuclear power plant is required to have an onsite technical support center to aid in the management of a nuclear power plant during an accident and a near-site emergency operations facility for coordination with State and local governments.

5. Public education programs on the nature of the hazard and appropriate actions during emergencies are required. Emergency preparedness information must be distributed annually to the public within about 10 miles of each nuclear power plant.

6. Periodic exercises are required to test utility, State, local and Federal emergency plans. Licensees are required to conduct an annual on-site emergency preparedness exercise and State and local governments are required to participate in such an exercise.

THE NRC AND FEMA EVALUATE THE EMERGENCY PREPAREDNESS CAPABILITIES OF THE UTILITIES AND THE STATE AND LOCAL GOVERNMENTS TO ENSURE A STATE OF READINESS TO RESPOND BY REVIEWING THE EMERGENCY PLANS, PERFORMING IMPLEMENTATION INSPECTIONS AND OBSERVING AND EVALUATING EXERCISES.

REP. BOGGS: GIVEN THE LACK OF CONTAINMENT BUILDINGS, HOW WOULD THE TIME AVAILABLE FOR EVACUATION OR OTHER EMERGENCY RESPONSE MEASURES COMPARE FOR THESE REACTORS AND FOR LICENSED REACTORS?

CHAIRMAN PALLADINO:

ACCORDING TO DOE, THE TIME AVAILABLE FOR EVACUATION OR OTHER EMERGENCY RESPONSE MEASURES AT DOE FACILITIES DEPENDS ON MANY THINGS, INCLUDING METEOROLOGICAL CONDITIONS AND DISTANCES. THE NOTIFICATION PROCESS AT THE SITES WHERE THE DEPARTMENT'S PRODUCTION REACTORS ARE LOCATED HAVE BEEN COORDINATED WITH SOUTH CAROLINA AND WASHINGTON. BECAUSE OF THE LARGE DISTANCES BETWEEN THE DOE PRODUCTION REACTORS AND THEIR SITE BOUNDARY, DOE BELIEVES NOTIFICATION TIMES ARE ADEQUATE.

THE PLANNING TIME FRAMES FOR ACCIDENTAL RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT IN EMERGENCY PLANS FOR NRC LICENSED REACTORS ARE BASED ON DESIGN BASIS ACCIDENT CONSIDERATIONS AND THE RESULTS OF CALCULATIONS REPORTED IN THE REACTOR SAFETY STUDY (WASH-1400). THE PLANNING BASIS FOR THE TIME DEPENDENCE OF A RELEASE IS EXPRESSED AS A RANGE OF TIME VALUES IN WHICH TO INITIATE PROTECTIVE ACTION. THIS RANGE OF VALUES PRIOR TO THE START OF A MAJOR RELEASE IS OF THE ORDER OF ONE-HALF HOUR TO SEVERAL HOURS FROM THE TIME OF THE INITIATING EVENT TO THE START OF AN ATMOSPHERIC RELEASE. THE SUBSEQUENT TIME PERIOD OVER WHICH

RADIOACTIVE MATERIAL MAY BE EXPECTED TO BE RELEASED IS OF THE ORDER OF ONE-HALF HOUR TO A FEW DAYS. THESE TIME FACTORS ARE BASED ON REACTORS WITH THE CONTAINMENT CHARACTERISTICS OF NRC LICENSED REACTORS AS ANALYZED IN THE REACTOR SAFETY STUDY. WHILE RECOGNIZING THAT THERE ARE SOME ACCIDENT SITUATIONS WHICH MAY BYPASS THE CONTAINMENT, THE CONTAINMENT PROVIDES SOME CAPABILITY FOR PREVENTING, DELAYING AND, IN CONJUNCTION WITH OTHER ENGINEERED SAFETY FEATURES, REMOVING RADIOACTIVE MATERIALS RELEASED FROM A REACTOR FOR SEVERE CORE MELTDOWN ACCIDENTS. THE EXTENT OF THIS CAPABILITY VARIES DEPENDING UPON INDIVIDUAL PLANT DESIGN, EQUIPMENT AND HUMAN PERFORMANCE AND THE SPECIFIC ACCIDENT SEQUENCE.

REP. AUCOIN: AS A REACTOR APPROACHES RETIREMENT AGE, DOES IT BECOME SAFER OR MORE ACCIDENT PRONE?

CHAIRMAN PALLADINO:

IN GENERAL, ANY SYSTEM OR COMPONENT, WITHOUT PROPER ATTENTION, WILL BECOME MORE PRONE TO FAILURE WITH AGE AND USE. FOR THIS REASON, THE NRC HAS CERTAIN REGULATIONS WHICH ENSURE THAT ALL REACTOR LICENSEES HAVE PROGRAMS TO INSPECT AND MAINTAIN SYSTEMS AND COMPONENTS NECESSARY TO ENSURE THE SAFETY OF THE PLANT. FOR EXAMPLE, INSERVICE INSPECTION IS REQUIRED ON IMPORTANT WELDS IN PIPING, AND VALVES WHICH PROVIDE ISOLATION BETWEEN HIGH AND LOW PRESSURE SYSTEMS ARE TESTED PERIODICALLY. LIKEWISE, PROTECTION SYSTEMS AND COMPONENTS (E.G., RELAYS) ARE TESTED PERIODICALLY AND EITHER REPAIRED OR REPLACED IF THEY DO NOT MEET PERFORMANCE SPECIFICATIONS.

THESE PROGRAMS ARE INTENDED TO ENSURE THAT U.S. PLANTS WILL NOT BECOME LESS SAFE AS THEY GET OLDER. HOWEVER, THE NRC IS CURRENTLY DEVELOPING PERFORMANCE INDICATORS FOR OPERATING NUCLEAR PLANTS WHICH MAY PROVIDE A BASIS FOR IDENTIFYING SYSTEMATIC DEGRADATION OF PLANT PERFORMANCE, INCLUDING INCREASED COMPONENT OR SYSTEM FAILURES THAT HAVE A POTENTIAL TO INITIATE ACCIDENTS. IN ADDITION, RECENT OPERATING EXPERIENCE INDICATES THAT THE SAFETY EFFECTS OF PLANT AGING DESERVES FURTHER ATTENTION BY THE INDUSTRY AND THE NRC.

MOREOVER, UNCERTAINTIES REMAIN IN CERTAIN AREAS OF PLANT AGING, AND THE NRC, AS WELL AS THE INDUSTRY, IS CONDUCTING RESEARCH PROGRAMS TO BETTER CHARACTERIZE THE AGING PROCESS AND ASSOCIATED UNCERTAINTIES.

REP. AUCOIN: WHAT IS THE DIFFERENCE BETWEEN A CONTAINMENT STRUCTURE AND A CONFINEMENT STRUCTURE?

CHAIRMAN PALLADINO:

A "CONTAINMENT STRUCTURE" ENCLOSES THE ENTIRE REACTOR AND REACTOR COOLANT SYSTEM AND SERVES AS ONE OF THE BARRIERS AGAINST RELEASE OF RADIOACTIVE FISSION PRODUCTS TO THE ENVIRONMENT UNDER POSTULATED DESIGN ACCIDENT CONDITIONS (I.E., PRESSURES AND TEMPERATURES).

THE WORD "CONFINEMENT STRUCTURE" IS NOT COMMONLY USED FOR DESCRIBING A SPECIFIC BUILDING OR STRUCTURE IN U.S. COMMERCIAL NUCLEAR POWER PLANTS. HOWEVER, IT IS CONCEPTUALIZED AS A CONTAINMENT STRUCTURE TO A LESSER DEGREE. IN REACTOR SAFETY TECHNOLOGY, THE WORD "CONFINEMENT" HAS MOST OFTEN BEEN APPLIED TO A SYSTEM (OR STRUCTURES) THAT PERMITS THE ESCAPE OF FILTERED GASES OR STEAM AFTER AN ACCIDENT IN A CONTROLLED MANNER.* IN THAT SENSE, THE SECONDARY CONTAINMENT STRUCTURE IN MANY LWR DESIGNS WHICH ENCLOSE THE CONTAINMENT STRUCTURE CAN BE TERMED AS A CONFINEMENT STRUCTURE.

REP. AUCOIN: WHY DOES THE NRC REQUIRE CONTAINMENT STRUCTURE FOR COMMERCIAL REACTORS, RATHER THAN CONFINEMENT STRUCTURE?

CHAIRMAN PALLADINO:

NRC REGULATIONS (10 CFR 50, APP. A, CRITERION 16) DO INDICATE THAT A CONTAINMENT IS REQUIRED AT ALL NUCLEAR POWER PLANTS.

THE INTENT OF THE NRC IN REQUIRING A CONTAINMENT STRUCTURE IS TO ENSURE AN ESSENTIALLY LEAK TIGHT BARRIER THAT PROVIDES EXTRA PROTECTION AGAINST EVEN SMALL RELEASES OF RADIOACTIVE FISSION PRODUCTS UNDER POSTULATED ACCIDENT CONDITIONS. THE POSTULATED ACCIDENT CONDITIONS INCLUDE THE CONSEQUENCES OF A RUPTURE OF THE LARGEST PRIMARY SYSTEM PIPE. THE RESULTING PRESSURE TRANSIENT CAN ONLY BE RETAINED BY A CONTAINMENT STRUCTURE. THE CONFINEMENT STRUCTURE WOULD NOT SERVE THE SAME PURPOSE.

REP. AUCOIN: HOW MUCH PRESSURE CAN A CONTAINMENT STRUCTURE WITHSTAND? HOW MUCH CAN A CONFINEMENT STRUCTURE WITHSTAND?

CHAIRMAN PALLADINO:

CONTAINMENT STRUCTURES FOR SOME PRESSURIZED WATER REACTORS (PWR) AND SOME BOILING WATER REACTORS (BWRS) ARE DESIGNED FOR BETWEEN 45 TO 60 PSIG PRESSURE. THE RECENT (POST TMI) ANALYTICAL AND NRC SPONSORED EXPERIMENTAL RESEARCH HAS INDICATED THE ACTUAL PRESSURE CAPACITIES OF THESE STRUCTURES TO BE 2½ TO 4 TIMES THE DESIGN PRESSURE. CONTAINMENT STRUCTURES OF LARGE SUPPRESSION TYPE DESIGNS (PWRS WITH ICE CONDENSERS AND BWR MK III) ARE DESIGNED FOR BETWEEN 10 TO 15 PSIG PRESSURE. RECENT ANALYSES HAVE INDICATED THE ACTUAL PRESSURE CAPACITIES OF THESE STRUCTURES TO BE 4 TO 6 TIMES THE DESIGN PRESSURE.

CONFINEMENT STRUCTURES CAN BE DESIGNED TO PERMIT LEAKAGE AT SPECIFIC PRESSURE - USUALLY IN THE RANGE OF 5 TO 10 PSIG.

COMMISSIONER ASSELSTINE ADDS THE FOLLOWING:

A MORE ACCURATE AND DEFENSIBLE RESPONSE IS THAT WE HAVE EXAMINED IN DETAIL A FEW CONTAINMENTS IN AN ATTEMPT TO DETERMINE THE ULTIMATE CAPACITY OF CONTAINMENTS. THOSE RESULTS SUGGEST THAT THE ACTUAL PRESSURE CAPACITIES OF THE EXAMINED CONTAINMENTS IS HIGHER, PERHAPS BY THE FACTORS MENTIONED IN THE COMMISSION'S RESPONSE, THAN THE DESIGN PRESSURE. HOWEVER, IT IS DIFFICULT AT THIS TIME TO MAKE A GENERIC STATEMENT OF HOW MUCH HIGHER BECAUSE INDIVIDUAL DESIGN AND CONSTRUCTION DETAILS CAN LEAD TO A LARGE VARIATION, PLANT-TO-PLANT, IN ULTIMATE CAPACITY UNDER SEVERE ACCIDENT CONDITIONS. WE HAVE A RESEARCH PROGRAM UNDERWAY TO DEVELOP ADDITIONAL INFORMATION ON THE ULTIMATE CAPACITIES.

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